Appendix I
Evaluation of Human Health Effects from Facility Accidents

This appendix presents the method and assumptions used for estimating potential impacts on, and risks to, individuals and the general public from exposure to releases of radioactive and hazardous chemical materials during hypothetical accidents at irradiation and processing facilities cited under the production alternatives described in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). The impacts of accidental radioactive material releases are given in Section I.1; the impacts of releases of hazardous chemicals, in Section I.2; and industrial accident impacts in Section I.3.

I.1 Radiological Accident Impacts on Human Health

The accidents considered in this NI PEIS for both the irradiation facilities and the processing facilities were based on a complete spectrum of accidents ranging from high-probability low-consequence events to extremely unlikely and incredible events. For this NI PEIS, a design-basis accident and a beyond-design-basis accident were specifically evaluated for each facility. More frequent events were specifically evaluated at the processing facilities because of the contribution to risk. These higher frequency events were not specifically evaluated for the irradiation facilities because they do not contribute to the risk (i.e., the risks of the design-basis accident and beyond-design-basis accidents are orders of magnitude higher than any more frequent event).

An extensive review of facility safety documentation (safety analysis reports, process hazard reviews, hazard analysis documents, and probabilistic risk analyses) was conducted. The review identified several accidents and their causes (initiating events). The initiating events reviewed included external events (e.g., airplane crashes, nearby explosions, fires), internal events (e.g., equipment failures, human error), natural phenomena (e.g., floods, tornadoes, earthquakes), and sabotage and terrorist activities. The review also determined that the only significant common-cause initiating event would be a catastrophic earthquake. In a common-cause event, the consequences from colocated facilities are summed. However, because of the low frequency of a catastrophic earthquake, the accidents evaluated in this NI PEIS bound the risks of a common-cause summation.

The accidents were grouped into one of four categories—anticipated occurrences, unlikely events, extremely unlikely events, or incredible events—based on the estimated frequency of occurrence. The accidents within each frequency category were examined to determine which accident(s) would result in the highest consequences (i.e., dose) and the highest risks (frequency × consequence). As a result, all other accident scenarios were screened from further consideration in this NI PEIS because the consequences and risks associated with those accidents would be lower than—or bounded by—the consequences and risks of the selected accidents.

The accident evaluation methodology ensures that all the facilities are treated on an equal basis. The analysis also considered facility-specific differences in design and mitigation features (e.g., filtration systems). Filtration efficiencies were obtained from facility safety reports, facility descriptions, and appropriate U.S. Department of Energy (DOE) and U.S. Nuclear Regulatory Commission (NRC) guidance.

I.1.1 Irradiation Facility Accident Scenario Selection and Description

A spectrum of potential accident scenarios was considered in this accident analysis assessment for the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee; the

Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho; a generic commercial light water reactor (CLWR); the Fast Flux Test Facility (FFTF) at the Hanford Site near Richland, Washington; low-energy and high-energy accelerators at a generic site; and a new research reactor at a generic site.

For each irradiation facility, a spectrum of accidents encompassing the full range of probabilities and consequences was considered for evaluation and inclusion in this NI PEIS. From the reactor final safety analysis reports, it was determined that only a few low-probability design-basis accidents or very low probability beyond-design-basis accidents contributed significantly to risk. Hence, only these events were specifically evaluated in this NI PEIS. In addition, handling accidents involving irradiated targets were also analyzed for HFIR, ATR, and FFTF. For the generic CLWR and the new research reactor, NRC guidance and published studies were used to determine appropriate design-basis and beyond-design-basis accidents. The specific guidance and studies used are presented in each of the following reactor analysis sections. The irradiation facilities analyses include the development of accident scenarios, transport of radioisotopes, and evaluation of health consequences, in addition to discussions of the methodologies used in these evaluations.

Accident	Frequency Range
Anticipated occurrences	1.0 – 0.01
Unlikely events	1×10^{-2} – 1×10^{-4}
Extremely unlikely events	1×10^{-4} – 1×10^{-6}
Incredible events	$<1\times10^{-6}$

Irradiation facility accident source terms include postulated neptunium-237 targets with a common spectrum of isotopes at the end of the plutonium-238 production cycle. The accident consequences were analyzed with end-of-cycle irradiated targets. Because of the radioisotope content, the end-of-cycle irradiated targets contribute most significantly to offsite consequences. **Table I–1** presents the inventory of target radioisotopes per gram of plutonium-238 produced.

Table I–1 Neptunium-237 Irradiated Target End-of-Cycle Nuclide Inventory (All Values Normalized to 1 Gram of Plutonium-238)

Isotope	Curies	Isotope	Curies
Cobalt-58	0.0	Tellurium-132	47.1
Cobalt-60	0.0	Iodine-131	32.5
Krypton-85	0.0202	Iodine-132	48.7
Krypton-85m	5.30	Iodine-133	65.0
Krypton-87	8.83	Iodine-134	69.0
Krypton-88	12.4	Iodine-135	60.8
Rubidium-86	0.00762	Xenon-133	61.3
Strontium-89	9.63	Xenon-135	7.69
Strontium-90	0.127	Cesium-134	0.159
Strontium-91	23.4	Cesium-136	0.92
Strontium-92	28.4	Cesium-137	0.375
Yttrium-90	0.128	Barium-139	54.1
Yttrium-91	13.2	Barium-140	45.1
Yttrium-92	28.7	Lanthanum-140	44.5
Yttrium-93	37.2	Lanthanum-141	51.3
Zirconium-95	24.7	Lanthanum-142	47.6
Zirconium-97	51.0	Cerium-141	35.0
Niobium-95	16.8	Cerium-143	42.5
Molybdenum-99	56.3	Cerium-144	7.13
Technetium-99m	50.1	Praseodymium-143	35.6
Ruthenium-103	42.5	Neodymium-147	17.1
Ruthenium-105	51.7	Neptunium-237	0.0036
Ruthenium-106	6.41	Neptunium-239	16.8
Rhodium-105	41.1	Plutonium-238	17
Antimony-127	4.44	Plutonium-239	0.00921
Antimony-129	13.5	Plutonium-240	0.00393
Tellurium-127	4.18	Plutonium-241	0.853
Tellurium-127m	0.243	Americium-241	0.0
Tellurium-129	12.9	Curium-242	0.0122
Tellurium-129m	1.39	Curium-244	0.0
Tellurium-131m	5.96	**Total**	**1,358.6**

Source: Schnitzler 1999.

The FFTF reactor, low-energy accelerator, and new research reactor accident source terms include medical, industrial, and research and development isotope targets. Projected radioisotope inventories for the target systems most likely to be considered for medical, industrial, and research and development isotope production are presented in **Table I–2**. These are maximum irradiated target inventories. The radium-226 target for actinium-227 production is the only target with a significantly radioactive target material. However, the dose due to the radium-226 target is insignificant compared with the dose due to the product isotopes. Therefore, the accident consequences were analyzed with the irradiated target products. Several of the isotope production targets generate substantial amounts of radioactive byproduct isotope in addition to the desired product. In these cases (gadolinium-153, actinium-227, and plutonium-238 production targets), the additional target inventory was included when calculating consequences.

Table I–2 Medical, Industrial, and Research and Development Isotope Irradiated Target Product Inventories

Product Isotope	Radioisotope	Target Inventory (curies)
Rapid Radioisotope Retrieval System		
Gold-198	Gold-198	132
Copper-64	Copper-64	1,300
Copper-67	Copper-67	6.26
Holmium-166	Holmium-166	58.9
Iodine-125	Iodine-125	2,530
Iodine-131	Iodine-131	307
Lutecium-177	Lutecium-177	0.519
Molybdenum-99	Molybdenum-99	1,680
Phosphorus-32	Phosphorus-32	39.1
Palladium-103	Palladium-103	1,340
Platinum-195m	Platinum-195m	168
Rhenium-186	Rhenium-186	4,350
Scandium-47	Scandium-47	29.6
Samarium-153	Samarium-153	70.7
Tin-117m	Tin-117m	48.5
Long-Term Irradiation Vehicle		
Cadmium-109	Cadmium-109	656
Gadolinium-153[a]	Gadolinium-153	1,100
Gadolinium-153	Europium-152	4,660
Gadolinium-153	Europium-152m	6.41×10^{4}
Gadolinium-153	Europium-154	1.55×10^{4}
Gadolinium-153	Europium-154m	2.20×10^{-4}
Gadolinium-153	Europium-155	3,540
Gadolinium-153	Europium-156	3.39×10^{5}
Gadolinium-153	Samarium-153	3.16×10^{4}
Iridium192	Iridium-192	3,570
Osmium-194	Osmium-194	2.20
Phosphorus-33	Phosphorus-33	76.2
Selenium-75	Selenium-75	17.9
Samarium-145	Samarium-145	11.8
Strontium-85	Strontium-85	2,160
Strontium-89	Strontium-89	156
Tungsten-188	Tungsten-188	5,810
Xenon-127	Xenon-127	7.26
Yttrium-91	Yttrium-91	17.8
Actinium-227[a]	Actinium-227	34.0
Actinium-227	Actinium-228	56.1
Actinium-227	Actinium-229	6.04×10^{-9}
Actinium-227	Radium-226	14.3
Actinium-227	Radium-227	4.23×10^{-7}
Actinium-227	Radium-228	0.00101
Actinium-227	Radium-229	5.00×10^{-14}
Actinium-227	Thorium-227	24.8
Actinium-227	Thorium-228	42.1

Product Isotope	Radioisotope	Target Inventory (curies)
Actinium-227	Thorium-229	8.63×10^{-4}
Actinium-227	Actinium-225	3.72×10^{-4}
Actinium-227	Astatine-217	3.72×10^{-4}
Actinium-227	Bismuth-210	0.109
Actinium-227	Bismuth-211	19.6
Actinium-227	Bismuth-212	24.6
Actinium-227	Bismuth-213	3.71×10^{-4}
Actinium-227	Bismuth-214	14.3
Actinium-227	Francium-221	3.72×10^{-4}
Actinium-227	Francium-223	1.40×10^{-5}
Actinium-227	Lead-209	3.69×10^{-4}
Actinium-227	Lead-210	0.118
Actinium-227	Lead-211	19.6
Actinium-227	Lead-212	38.4
Actinium-227	Lead-214	14.3
Actinium-227	Polonium-210	0.106
Actinium-227	Polonium-211	0.0535
Actinium-227	Polonium-212	24.6
Actinium-227	Polonium-213	3.63×10^{-4}
Actinium-227	Polonium-214	14.3
Actinium-227	Polonium-215	19.6
Actinium-227	Polonium-216	38.8
Actinium-227	Polonium-218	14.3
Actinium-227	Radium-223	19.6
Actinium-227	Radium-224	38.8
Actinium-227	Radium-225	5.46×10^{-4}
Actinium-227	Radon-217	4.46×10^{-8}
Actinium 227	Radon-219	19.6
Actinium-227	Radon-220	38.8
Actinium-227	Radon-222	14.3
Actinium-227	Thallium-207	19.6
Actinium-227	Thallium-208	8.83
Actinium-227	Thallium-209	8.16×10^{-6}

a. The gadolinium-153 and actinium-227 production targets include radioactive byproducts.
Source: BWHC 1999.

I.1.1.1 Advanced Test Reactor

ATR would generate 5 kilograms (11 pounds) per year of plutonium-238 in support of Alternative 2, Options 1 through 3, and 3 kilograms (6.6 pounds) per year of plutonium-238 in support of Alternative 2, Options 7 through 9. On average, ATR has seven refueling outages per year. ATR accident analyses assumed that one-seventh of the annual plutonium-238 production would be harvested at each refueling outage and an equal amount of plutonium-238 would remain in the core in targets that were not ready to be harvested. The accident analyses postulated that the plutonium-238 at risk in targets during ATR accidents is 857 grams (1.89 pounds) for the annual production rate of 3 kilograms (6.6 pounds) per year and 1,429 grams (3.144 pounds) for the annual production rate of 5 kilograms (11 pounds) per year.

I.1.1.1.1 Design-Basis Accident

The *ATR Upgraded Final Safety Analysis Report* (LMIT 1998) stated that seven design-basis accidents would provide the greatest challenge to the engineered safety features of ATR. These accidents and the affected engineered safety systems are summarized in **Table I–3**.

Table I–3 ATR Engineered Safety Feature Design-Basis Accidents

Accident Sequence	Engineered Safety System
3-inch diameter opening in the primary coolant system due to an opening of a drain valve, relief valve, or vent valve	Emergency firewater injection system
Experiment loop piping failure	Radiation monitoring and seal system
Long-term complete loss of flow or complete loss of heat sink	Primary coolant overpressure relief and vent systems and emergency firewater injection system
Opening of flow control butterfly valve to full open	Primary pump shutoff system
Loss of primary coolant system pressure control (loss of instrument air)	Pressurizing pumps and gland seal pumps shutoff system
Loss of pressure control of primary coolant system and failure of the pressurizing pumps and gland seal pumps shutoff system	Primary coolant overpressure relief system
Loss of primary coolant system inventory during depressurized and outage operations when irradiated fuel elements are in the reactor vessel	Vessel level alarm system

Source: LMIT 1998.

The accident sequences listed in the table do not lead to core damage and do not have the potential to damage appropriately designed neptunium-237 targets being irradiated in the core.

I.1.1.1.2 Severe Reactor Accident

The large-break loss-of-coolant accident postulated for ATR is a severe reactor accident. This event would result in a decrease in the primary coolant inventory of ATR. As treated in the *ATR Upgraded Final Safety Analysis Report*, the large-break loss-of-coolant accident is a limiting accident compared with other initiating events because 100 percent core damage is estimated to occur. The probability for the occurrence of an ATR large-break loss-of-coolant accident is on the order of 1×10^{-4} per year.

The radiological analysis of the large-break loss-of-coolant accident shows that an ATR core inventory of 1.11 gigacuries at reactor scram conditions releases an available source term of 175 megacuries (LMIT 1998). The emergency firewater injection system is assumed to pump water through the break into confinement, until shutoff level is reached, about 33 hours after the break. Within that period, about 65 percent of the available source term, or 113 megacuries, will have been released as the early release source term. Following the termination of emergency firewater injection system flow at 33 hours, the confinement leak rate is assumed to drop to the design value of 10 percent per day, resulting in a release of the remaining 62 megacuries as the late-release source term, ending about 85 hours after the loss-of-coolant accident. Consequently, the total release duration for the large-break loss-of-coolant accident is 118.5 hours, or the sum of 33.3 hours for the early-release source term and 85.2 hours for the late-release source term.

The core inventories and environmental releases for the three possible plutonium-238 production rates (0, 3, or 5 kilograms per year) are presented in **Table I–4**. The core inventory was based on a maximum design power level of 250 megawatts.

Table I–4 ATR Large-Break Loss-of-Coolant Accident Source Terms

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate			Environmental Release (curies) Versus Plutonium-238 Production Rate		
	0 kilograms per year	3 kilograms per year	5 kilograms per year	0 kilograms per year	3 kilograms per year	5 kilograms per year
Krypton-85	5,900	6,000	6,000	5,900	6,000	6,000
Krypton-85m	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}
Krypton-87	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}
Krypton-88	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}
Rubidium-86	3,900	3,900	3,900	1,900	1,900	1,900
Strontium-89	5.6×10^{6}	5.6×10^{6}	5.6×10^{6}	3.4×10^{5}	3.4×10^{5}	3.4×10^{5}
Strontium-90	4.7×10^{4}	4.7×10^{4}	4.7×10^{4}	2,800	2,800	2,800
Strontium-91	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.1×10^{5}	7.1×10^{5}	7.1×10^{5}
Strontium-92	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.2×10^{5}	7.3×10^{5}	7.3×10^{5}
Yttrium-90	4.9×10^{4}	4.9×10^{4}	4.9×10^{4}	200	200	200
Yttrium-91	6.2×10^{6}	6.2×10^{6}	6.2×10^{6}	2.5×10^{4}	2.5×10^{4}	2.5×10^{4}
Yttrium-92	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.9×10^{4}	4.9×10^{4}
Yttrium-93	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	5.2×10^{4}	5.2×10^{4}	5.2×10^{4}
Zirconium-95	6.4×10^{6}	6.4×10^{6}	6.4×10^{6}	2.6×10^{4}	2.6×10^{4}	2.6×10^{4}
Zirconium-97	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Niobium-95	2.9×10^{6}	2.9×10^{6}	2.9×10^{6}	1.1×10^{4}	1.2×10^{4}	1.2×10^{4}
Molybdenum-99	1.2×10^{7}	1.3×10^{7}	1.3×10^{7}	2.5×10^{5}	2.5×10^{5}	2.5×10^{5}
Technetium-99m	1.1×10^{7}	1.1×10^{7}	1.1×10^{7}	2.2×10^{5}	2.3×10^{5}	2.3×10^{5}
Ruthenium-103	4.3×10^{6}	4.3×10^{6}	4.3×10^{6}	8.6×10^{4}	8.6×10^{4}	8.7×10^{4}
Ruthenium-105	2.2×10^{6}	2.3×10^{6}	2.3×10^{6}	4.4×10^{4}	4.5×10^{4}	4.6×10^{4}
Ruthenium-106	9.8×10^{4}	1.0×10^{5}	1.1×10^{5}	2,000	2,100	2,100
Rhodium-105	1.5×10^{6}	1.5×10^{6}	1.5×10^{6}	3.0×10^{4}	3.0×10^{4}	3.1×10^{4}
Antimony-127	3.4×10^{5}	3.4×10^{5}	3.4×10^{5}	1.0×10^{5}	1.0×10^{5}	1.0×10^{5}
Antimony-129	1.4×10^{6}	1.4×10^{6}	1.4×10^{6}	4.2×10^{5}	4.3×10^{5}	4.3×10^{5}
Tellurium-127	3.2×10^{5}	3.2×10^{5}	3.2×10^{5}	0	0	0
Tellurium-127m	1.4×10^{4}	1.4×10^{4}	1.4×10^{4}	0	0	0
Tellurium-129	1.4×10^{6}	1.4×10^{6}	1.4×10^{6}	0	0	0
Tellurium-129m	1.5×10^{5}	1.5×10^{5}	1.5×10^{5}	0	0	0
Tellurium-131	5.3×10^{6}	5.3×10^{6}	5.3×10^{6}	0	0	0
Tellurium-131m	7.6×10^{5}	7.6×10^{5}	7.7×10^{5}	0	0	0
Tellurium-132	8.9×10^{6}	9.0×10^{6}	9.0×10^{6}	0	0	0
Iodine-131	6.0×10^{6}	6.0×10^{6}	6.0×10^{6}	3.2×10^{5}	3.2×10^{5}	3.2×10^{5}
Iodine-132	9.1×10^{6}	9.1×10^{6}	9.1×10^{6}	4.8×10^{5}	4.8×10^{5}	4.8×10^{5}
Iodine-133	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	7.3×10^{5}	7.3×10^{5}	7.3×10^{5}
Iodine-134	1.5×10^{7}	1.5×10^{7}	1.6×10^{7}	8.2×10^{5}	8.2×10^{5}	8.2×10^{5}
Iodine-135	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	6.8×10^{5}	6.8×10^{5}	6.8×10^{5}
Xenon-133	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}
Xenon-135	4.7×10^{5}	4.8×10^{5}	4.8×10^{5}	4.7×10^{5}	4.8×10^{5}	4.8×10^{5}
Cesium-134	3.8×10^{4}	3.8×10^{4}	3.8×10^{4}	0	0	0
Cesium-136	2.7×10^{4}	2.8×10^{4}	2.8×10^{4}	0	0	0
Cesium-137	4.8×10^{4}	4.9×10^{4}	4.9×10^{4}	0	0	0
Barium-139	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	7.8×10^{5}	7.9×10^{5}	7.9×10^{5}
Barium-140	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.5×10^{5}	7.5×10^{5}	7.5×10^{5}

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate			Environmental Release (curies) Versus Plutonium-238 Production Rate		
	0 kilograms per year	3 kilograms per year	5 kilograms per year	0 kilograms per year	3 kilograms per year	5 kilograms per year
Lanthanum-140	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	5.0×10^{4}	5.0×10^{4}	5.0×10^{4}
Lanthanum-141	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Lanthanum-142	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Cerium-141	8.8×10^{6}	8.8×10^{6}	8.8×10^{6}	3.5×10^{4}	3.5×10^{4}	3.5×10^{4}
Cerium-143	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.9×10^{4}	4.9×10^{4}	4.9×10^{4}
Cerium-144	1.5×10^{6}	1.5×10^{6}	1.5×10^{6}	6,200	6,200	6,200
Praseodymium-143	1.1×10^{7}	1.1×10^{7}	1.1×10^{7}	4.5×10^{4}	4.5×10^{4}	4.5×10^{4}
Neodymium-147	4.4×10^{6}	4.4×10^{6}	4.4×10^{6}	1.8×10^{4}	1.8×10^{4}	1.8×10^{4}
Neptunium-237	8.5×10^{-2}	3.2	5.2	3.4×10^{-4}	1.3×10^{-2}	2.1×10^{-2}
Neptunium-239	3.7×10^{6}	3.8×10^{6}	3.8×10^{6}	1.5×10^{4}	1.5×10^{4}	1.5×10^{4}
Plutonium-238	170	1.5×10^{4}	2.4×10^{4}	0.69	59	97
Plutonium-239	6.5	14	20	0.026	0.058	0.079
Plutonium-240	4.1	7.5	9.7	0.016	0.030	0.039
Plutonium-241	1,500	2,300	2,800	6.1	9.1	11
Americium-241	0.088	0.088	0.088	3.5×10^{-4}	3.5×10^{-4}	3.5×10^{-4}
Curium-242	15	25	32	0.059	0.10	0.13
Curium-244	1.3	1.3	1.3	0.0052	0.0052	0.0052

Source: LMIT 1998 and Schnitzler 1999.

The ATR core inventory and release fractions were obtained from the *ATR Upgraded Final Safety Analysis Report* which provides the end-of-cycle core inventory for several hundred isotopes (LMIT 1998). These isotopes were screened and reduced to those that contribute to human health effects.

I.1.1.1.3 Neptunium-237 Target-Handling Accident

The neptunium-237 target-handling accident scenario postulates the maximum amount of targets in the storage pool. A drop sufficient to damage the entire neptunium-237 target inventory is assumed. This accident is assumed to have a likelihood of occurrence of 0.001 per year.

For the purposes of this analysis, the following assumptions are made for the target-handling accident. The fuel-clad gap contains 10 percent of the fission product gases and iodine (NRC 1978). One-hundred percent of the noble gases and tritium gas in the fuel-clad gap is released to the environment through the reactor building exhaust system. This results in an overall release fraction of 0.1 for the noble gases and tritium. Twenty-five percent of the iodine in the fuel-clad gap is released from the fuel assembly, and 90 percent of the released iodine is absorbed in the reactor pool. The remaining iodine is released to the environment through the reactor building exhaust system. The exhaust system charcoal filter is assumed to remove 99 percent of the iodine (NRC 1978). This results in an overall release fraction of 2.5×10^{-5} ($0.1 \times 0.25 \times 0.1 \times 0.01 = 2.5\times10^{-5}$) for the iodine. These assumptions result in the source terms shown in **Table I–5** for the 3- and 5-kilogram-per-year (6.6- and 11-pounds-per-year) production rates.

Table I–5 ATR Neptunium-237 Target-Handling Accident Source Terms

Isotope	Environmental Release (curies) Versus Plutonium-238 Production Rate	
	3 kilograms per year	5 kilograms per year
Hydrogen-3	0.207	0.344
Krypton-85	1.73	2.89
Krypton-85m	454	757
Krypton-87	757	1,260
Krypton-88	1,060	1,770
Iodine-131	0.698	1.16
Iodine-132	1.04	1.74
Iodine-133	1.39	2.07
Iodine-134	1.48	2.47
Iodine-135	1.30	2.17
Xenon-133	5,250	8,760
Xenon-135	659	1,100

Source: Calculated results.

I.1.1.1.4 Meteorological Data

Meteorological characteristics of the ATR site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at INEEL.

I.1.1.1.5 Population Data

The population distribution surrounding ATR is based on the 1990 Census of Population and Housing (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.1.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for ATR accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on U.S. Environmental Protection Agency (EPA) Protective Action Guides.

I.1.1.2 High Flux Isotope Reactor Accident Analyses

HFIR would generate 2 kilograms (4.4 pounds) per year of plutonium-238 in support of Alternative 2, Options 7 through 9. On average, HFIR has 11 refueling outages per year. HFIR accident analyses assumed that one-eleventh of the annual plutonium-238 production would be harvested at each refueling outage and an equal amount of plutonium-238 would remain in the core in targets that were not ready to be harvested. The accident analyses postulated that the plutonium-238 at risk in targets during HFIR accidents is 364 grams (0.80 pound).

I.1.1.2.1 Design-Basis Accident

The *HFIR Safety Analysis Report* (LMER 1998) detailed numerous small-break loss-of-coolant accidents. The worst-case scenario is a 2-inch (5-centimeter) break at the reactor vessel. The primary flow drops sharply in the first few seconds after the break before recovering at about one-fourth of its normal value. However, the primary coolant system fluid remains subcooled throughout the event, and there is considerable margin to critical heat flux. This is the maximum tolerable break short of fuel damage and nonrecoverable flow. It also represents the largest break size that still has a frequency of occurrence greater than 1×10^{-4} per year.

No reactor fuel or target rods fail as a result of the worst-case small-break loss-of-coolant accident.

I.1.1.2.2 Severe Reactor Accident

The large-break loss-of-coolant accident is the limiting severe reactor accident at HFIR. Two large-break loss-of-coolant accidents were evaluated in the *HFIR Safety Analysis Report* (LMER 1998). Both accidents involve breaks in the primary coolant system piping. The first is a double-ended guillotine break of the cold leg in the reactor pool, in which the reactor coolant is retained inside confinement. The second is a double-ended guillotine break of a primary coolant pump discharge line in a heat exchanger cell. The consequences of a large-break loss-of-coolant accident in the heat exchanger cell are bounded by those resulting from a large-break loss-of-coolant accident in the reactor pool. Therefore, the large-break loss-of-coolant accident in the reactor pool was chosen for analysis in this NI PEIS.

The large-break loss-of-coolant accident in the reactor pool assumes that 100 percent of the core melts. Equipment in service at the beginning of the accident is assumed to operate for the duration of the accident. This equipment includes the special building or confinement hot-exhaust system, which is designed to filter out airborne particulate activity from the HFIR building.

The *HFIR Safety Analysis Report* (LMER 1998) states that 100 percent of noble gases and 1 percent of iodines are released to the environment. The accident scenario presented in the facility safety analysis report assumes that the primary coolant piping breaks in the reactor pool. Therefore, even though the primary coolant piping inventory is lost, the core remains covered with water. Because of this assumption, only noble gases and iodine are assumed to be released to the environment. This differs from other reactors in the assumption that no other radioisotopes are released. For most reactors, a severe loss-of-coolant accident results in an uncovered core, leading to a fractional release of all isotopes.

The accident source term is presented in **Table I–6** for the two possible HFIR core configurations.

Table I–6 HFIR Large-Break Loss-of-Coolant Accident Source Term

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate		Environmental Release (curies) Versus Plutonium-238 Production Rate	
	0 kilograms per year	2 kilograms per year	0 kilograms per year	2 kilograms per year
Krypton-85	800	810	800	810
Krypton-85m	8.8×10^{5}	8.8×10^{5}	8.8×10^{5}	8.8×10^{5}
Krypton-87	1.8×10^{6}	1.8×10^{6}	1.8×10^{6}	1.8×10^{6}
Krypton-88	$2\ .6\times10^{6}$	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}
Rubidium-86	130	130	0	0
Strontium-89	9.5×10^{5}	9.6×10^{5}	0	0
Strontium-90	6,500	6,600	0	0
Strontium-91	4.1×10^{6}	4.1×10^{6}	0	0
Yttrium-90	5,600	5,600	0	0
Yttrium-91	1.0×10^{6}	1.0×10^{6}	0	0
Zirconium-95	1.0×10^{6}	1.0×10^{6}	0	0
Zirconium-97	4.1×10^{6}	4.1×10^{6}	0	0
Niobium-95	2.2×10^{5}	2.3×10^{5}	0	0
Molybdenum-99	4.2×10^{6}	4.3×10^{6}	0	0
Technetium-99m	3.9×10^{6}	3.9×10^{6}	0	0
Ruthenium-103	7.7×10^{5}	7.8×10^{5}	0	0
Ruthenium-105	7.2×10^{5}	7.4×10^{5}	0	0
Ruthenium-106	1.3×10^{4}	1.5×10^{4}	0	0
Rhodium-105	5.8×10^{5}	5.9×10^{5}	0	0
Antimony-127	1.1×10^{5}	1.1×10^{5}	0	0
Antimony-129	4.8×10^{5}	4.9×10^{5}	0	0
Tellurium-127	9.6×10^{4}	9.8×10^{4}	0	0
Tellurium-127m	1,700	1,800	0	0
Tellurium-129	4.5×10^{5}	4.5×10^{5}	0	0
Tellurium-129m	2.8×10^{4}	2.8×10^{4}	0	0
Tellurium-131m	2.6×10^{5}	2.6×10^{5}	0	0
Tellurium-132	3.0×10^{6}	3.0×10^{6}	0	0
Iodine-131	1.7×10^{6}	1.7×10^{6}	1.7×10^{4}	1.7×10^{4}
Iodine-132	3.0×10^{6}	3.0×10^{6}	3.0×10^{4}	3.0×10^{4}
Iodine-133	4.6×10^{6}	4.6×10^{6}	4.6×10^{4}	4.6×10^{4}
Iodine-134	5.4×10^{6}	5.4×10^{6}	5.4×10^{4}	5.4×10^{4}
Iodine-135	4.4×10^{6}	4.4×10^{6}	4.4×10^{4}	4.4×10^{4}
Xenon-133	4.6×10^{6}	4.6×10^{6}	9.2×10^{6}	9.2×10^{6}
Xenon-135	1.5×10^{5}	1.5×10^{5}	3.5×10^{6}	3.5×10^{6}
Cesium-134	440	500	0	0
Cesium-136	4,000	4,300	0	0
Cesium-137	6,600	6,700	0	0
Cerium-141	1.6×10^{6}	1.6×10^{6}	0	0
Cerium-143	4.1×10^{6}	4.2×10^{6}	0	0
Cerium-144	2.2×10^{5}	2.2×10^{5}	0	0
Barium-140	3.2×10^{6}	3.2×10^{6}	0	0
Lanthanum-140	3.1×10^{6}	3.1×10^{6}	0	0
Praseodymium-143	2.8×10^{6}	2.8×10^{6}	0	0
Neodymium-147	1.3×10^{6}	1.3×10^{6}	0	0
Neptunium-237	0	1.3	0	0
Neptunium-239	2.9×10^{5}	3.0×10^{5}	0	0

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate		Environmental Release (curies) Versus Plutonium-238 Production Rate	
	0 kilograms per year	2 kilograms per year	0 kilograms per year	2 kilograms per year
Plutonium-238	0.32	6,200	0	0
Plutonium-239	0.38	3.7	0	0
Plutonium-240	0.055	1.5	0	0
Plutonium-241	1.1	310	0	0
Americium-241	2.4×10^{-5}	2.4×10^{-5}	0	0
Curium-242	4.6×10^{-4}	4.4	0	0
Curium-244	9.9×10^{-7}	9.9×10^{-7}	0	0

Source: Rothrock 1999; Schnitzler 1999; Wham 1999.

I.1.1.2.3 Neptunium-237 Target-Handling Accident

The neptunium-237 target-handling accident scenario postulates the maximum number of targets in the storage pool. A drop sufficient to damage the entire neptunium-237 target inventory is assumed. This accident is assumed to have a likelihood of occurrence of 0.001 per year. The accident assumptions are described in Section I.1.1.1.3. These assumptions result in the source terms, shown in **Table I–7**, for a 2-kilograms-per-year (4.4-pounds-per-year) production rate.

Table I–7 HFIR Neptunium-237 Target-Handling Accident Source Term

Isotope	Environmental Release[a] (curies)
Hydrogen-3	0.0877
Krypton-85	0.735
Krypton-85m	193
Krypton-87	321
Krypton-88	451
Iodine-131	0.295
Iodine-132	0.443
Iodine-133	0.593
Iodine-134	0.628
Iodine-135	0.553
Xenon-133	2,230
Xenon-135	280

a. Based on a 2-kilogram-per-year plutonium-238 production rate.
Source: Calculated results.

I.1.1.2.4 Meteorological Data

Meteorological characteristics of the HFIR site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at ORNL.

I.1.1.2.5 Population Data

The population distribution surrounding HFIR is based on the 1990 census (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.2.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for HFIR accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.3 Commercial Light Water Reactor

The CLWR would generate 5 kilograms (11 pounds) per year of plutonium-238 in support of Option 4, 5, or 6 of Alternative 2. On average, CLWR has one refueling outage every 18 months. The accident analysis assumes that 100 percent of the targets in the reactor core would be harvested at each refueling outage. The analysis postulates that the plutonium-238 at risk in targets during CLWR operation is 7.5 kilograms (16.5 pounds).

The analysis is based primarily on NUREG/CR-6295 (Davis 1997). NUREG/CR-6295 provides simplified design-basis and severe-accident source terms and generic site parameters based on the risk insights of NUREG-1150 (NRC 1990). These simplified source terms and generic parameters are used to analyze accidents for the current core for a baseline impact and with the proposed neptunium-237 targets to determine the incremental impact of plutonium-238 production. Core damage and containment failure frequencies were updated using more recent risk insights from the Individual Plant Examination (IPE) database (NRC 1997).

I.1.1.3.1 Core Inventories

After a review of NUREG/CR-6295, the 3,800 megawatts-thermal pressurized-water-reactor accident release fractions were chosen for this analysis. This reactor has the highest energy level and the consequences result in the highest risk of the reactors analyzed in NUREG/CR-6295. The MELCOR Accident Consequence Code System (MACCS2) documentation provides a typical end-of-cycle core inventory for a 3,412 megawatts-thermal pressurized-water-reactor. This power level was selected for the analysis because only 5 of the 73 currently operating pressurized water reactors have higher power levels, and 19 have a power level of 3,411 megawatts-thermal.

Table I–8 provides inventories for the current core configuration, the target inventory, and the core-containing targets. The end-of-cycle inventories provide bounding source terms which lead to maximum consequences. The calculation conservatively assumes that the targets are additions to the core and not replacements for some fuel rods. Replacing some burned fuel rods with targets would lower the core activity, perhaps below that without targets. As noted in the total activities line of the table, there is very little difference (approximately 0.16 percent) between the current core at 6.37×10^{9} curies versus 6.38×10^{9} curies for the current core plus the targets.

Table I–8 Core Inventories Based on a Target Maximum Core Loading of 7.5 Kilograms

Isotope	Core Inventory (curies)[a]	Target Inventory (curies)[b]	Core + Target Inventory (curies)
Cobalt-58	8.71×10^{5}	0.00	8.71×10^{5}
Cobalt-60	6.66×10^{5}	0.00	6.66×10^{5}
Krypton-85	6.69×10^{5}	152	6.69×10^{5}
Krypton-85m	3.13×10^{7}	3.98×10^{4}	3.14×10^{7}
Krypton-87	5.72×10^{7}	6.62×10^{4}	5.73×10^{7}
Krypton-88	7.74×10^{7}	9.30×10^{4}	7.75×10^{7}
Rubidium-86	5.10×10^{4}	57.2	5.11×10^{4}
Strontium-89	9.70×10^{7}	7.22×10^{4}	9.71×10^{7}
Strontium-90	5.24×10^{6}	953	5.24×10^{6}
Strontium-91	1.25×10^{8}	1.76×10^{5}	1.25×10^{8}
Strontium-92	1.30×10^{8}	2.13×10^{5}	1.30×10^{8}
Yttrium-90	5.62×10^{6}	960	5.62×10^{6}
Yttrium-91	1.18×10^{8}	9.90×10^{4}	1.18×10^{8}
Yttrium-92	1.30×10^{8}	2.15×10^{5}	1.31×10^{8}
Yttrium-93	1.47×10^{8}	2.79×10^{5}	1.48×10^{8}
Zirconium-95	1.49×10^{8}	1.85×10^{5}	1.50×10^{8}
Zirconium-97	1.56×10^{8}	3.83×10^{5}	1.56×10^{8}
Niobium-95	1.41×10^{8}	1.26×10^{5}	1.41×10^{8}
Molybdenum-99	1.65×10^{8}	4.22×10^{5}	1.65×10^{8}
Technetium-99m	1.42×10^{8}	3.76×10^{5}	1.43×10^{8}
Ruthenium-103	1.23×10^{8}	3.19×10^{5}	1.23×10^{8}
Ruthenium-105	7.98×10^{7}	3.88×10^{5}	8.02×10^{7}
Ruthenium-106	2.79×10^{7}	4.81×10^{4}	2.79×10^{7}
Rhodium-105	5.53×10^{7}	3.08×10^{5}	5.56×10^{7}
Antimony-127	7.53×10^{6}	3.33×10^{4}	7.57×10^{6}
Antimony-129	2.67×10^{7}	1.01×10^{5}	2.68×10^{7}
Tellurium-127	7.28×10^{6}	3.14×10^{4}	7.31×10^{6}
Tellurium-127m	9.63×10^{5}	1,820	9.65×10^{5}
Tellurium-129	2.50×10^{7}	9.68×10^{4}	2.51×10^{7}
Tellurium-129m	6.60×10^{6}	1.04×10^{4}	6.61×10^{6}
Tellurium-131m	1.26×10^{7}	4.47×10^{4}	1.27×10^{7}
Tellurium-132	1.26×10^{8}	3.53×10^{5}	1.26×10^{8}
Iodine-131	8.66×10^{7}	2.44×10^{5}	8.69×10^{7}
Iodine-132	1.28×10^{8}	3.65×10^{5}	1.28×10^{8}
Iodine-133	1.83×10^{8}	4.88×10^{5}	1.84×10^{8}
Iodine-134	2.01×10^{8}	5.18×10^{5}	2.02×10^{8}
Iodine-135	1.73×10^{8}	4.56×10^{5}	1.73×10^{8}
Xenon-133	1.83×10^{8}	4.60×10^{5}	1.84×10^{8}
Xenon-135	3.44×10^{7}	5.77×10^{4}	3.45×10^{7}
Cesium-134	1.17×10^{7}	1,190	1.17×10^{7}
Cesium-136	3.56×10^{6}	6,900	3.56×10^{6}
Cesium-137	6.53×10^{6}	2,810	6.54×10^{6}
Barium-139	1.70×10^{8}	4.06×10^{5}	1.70×10^{8}
Barium-140	1.68×10^{8}	3.38×10^{5}	1.68×10^{8}
Lanthanum-140	1.72×10^{8}	3.34×10^{5}	1.72×10^{8}
Lanthanum-141	1.57×10^{8}	3.85×10^{5}	1.58×10^{8}
Lanthanum-142	1.52×10^{8}	3.57×10^{5}	1.52×10^{8}
Cerium-141	1.53×10^{8}	2.63×10^{5}	1.53×10^{8}
Cerium-143	1.48×10^{8}	3.19×10^{5}	1.49×10^{8}
Cerium-144	9.20×10^{7}	5.35×10^{4}	9.21×10^{7}

Isotope	Core Inventory (curies)[a]	Target Inventory (curies)[b]	Core + Target Inventory (curies)
Praseodymium-143	1.46×10^{8}	2.67×10^{5}	1.46×10^{8}
Neodymium-147	6.52×10^{7}	1.28×10^{5}	6.53×10^{7}
Neptunium-239	1.75×10^{9}	1.26×10^{5}	1.75×10^{9}
Plutonium-238	9.90×10^{4}	1.27×10^{5}	2.26×10^{5}
Plutonium-239	2.23×10^{4}	69.1	2.24×10^{4}
Plutonium-240	2.82×10^{4}	29.5	2.82×10^{4}
Plutonium-241	4.74×10^{6}	6,400	4.75×10^{6}
Americium-241	3,130	0.00	3,130
Curium-242	1.20×10^{6}	91.5	1.20×10^{6}
Curium-244	7.02×10^{4}	0.00	7.02×10^{4}
Totals	6.37×10^{9}	1.02×10^{7}	6.38×10^{9}

a. Chanin et al. 1990; inventory converted from becquerels (Bq) to curies (Ci); 3.7×10^{10} Bq = 1 Ci.
b. Schnitzler 1999.

I.1.1.3.2 Meteorological Data

According to NUREG/CR-6295 (Davis 1997), the Sandia Siting Study evaluated data from 29 National Weather Service sites representing the nation's meteorological conditions. The 29 sites were compared to determine which site best represents the nation's meteorological conditions. It was determined that the site with the least deviation from the mean is the one at Omaha, Nebraska. Another comparison of the 29 sites indicated that the mean mixing height is 1.5 kilometers (0.93 mile). The mean meteorological data used in the NI PEIS analysis are a composite of the Omaha meteorological conditions and the mean mixing height.

I.1.1.3.3 Population Data

To be as generic as possible, the population around the plant was assumed to be uniformly distributed. The analysis was performed for a population density of 100 persons per square mile (38.6 persons per square kilometer) from 0 to 10 miles (representing the median population density for all pressurized water reactors) and 200 persons per square mile (77.2 persons per square kilometer) from 10 to 50 miles (representing an average population density beyond 10 miles). The exclusion area boundary was assumed to be 640 meters (0.4 mile) from the reactor.

I.1.1.3.4 Evacuation Information

Consistent with NUREG-1150, this analysis assumes that 99.5 percent of the population within the 16.1-kilometer (10-mile) emergency planning zone participates in an evacuation. It was also assumed that the 0.5 percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hours after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with EPA guidelines. Mean evacuation time and speed were based on the average of the five NUREG-1150 plants. This results in an evacuation delay time of 1.9 hours and an evacuation speed of 9.3 kilometers (5.8 miles) per hour.

I.1.1.3.5 Design-Basis Accident

Design-basis events are defined by the American Nuclear Society as Condition IV occurrences or limiting faults. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial radioactive material. These are the most serious events which must be designed against and represent limiting design cases.

A realistic design-basis large-break loss-of-coolant accident was chosen for evaluation because it is the limiting design-basis accident at pressurized water reactor plants. The large-break loss-of-coolant accident is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following a postulated double-ended rupture of a reactor coolant pipe, the emergency core cooling system keeps cladding temperatures well below melting, ensuring that the core remains intact and in a coolable geometry. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur for the design-basis loss-of-coolant accident, a postulated gross release of fission products is evaluated in accordance with NRC accident analysis guidelines (AEC 1974). The only postulated mechanism for such a release would be a number of simultaneous and extended failures in the engineered safety feature systems, producing severe physical degradation of core geometry and partial melting of the fuel.

The realistic large-break loss-of-coolant accident release characteristics, obtained from NUREG/CR-6295, are described by the release height, timing, duration, and heat content of the plume; the fraction of each isotope group released; and the warning time (time when offsite officials are warned that an emergency response should be initiated.) **Tables I–9 and I–10** provide the release parameters for the realistic large-break loss-of-coolant accident.

Table I–9 Design-Basis Accident Release Characteristics

Accident Scenario	Scenario Frequency	Elevation of Release (m)	Energy of Release (W)	Warning Time (hr)	Time of Release (hr)	Duration of Release (hr)
Large-break loss-of-coolant accident[a]	4.65×10^{-5}	0	0.0	5.0	6.0	10.0
					16.0	

a. The accident is represented by two separate releases.
Key: hr, hour; m, meter; W, watts.
Source: Davis 1997.

Table I–10 Design-Basis Accident Release Fractions

Release Category	Release Fractions by Isotope								
	Kr, Xe	I	Cs, Rb	Sb, Te	Sr	Co, Mo, Rh, Ru, Tc	Am, Cm, La, Nb, Nd, Pr, Y, Zr	Ce, Np, Pu	Ba
Large-break loss-of-coolant accident[a]	2.5×10^{-3}	1.5×10^{-5}	1.2×10^{-8}	7.5×10^{-9}	2.5×10^{-9}	2.0×10^{-10}	3.0×10^{-10}	4.0×10^{-10}	2.5×10^{-9}
	2.5×10^{-3}	1.5×10^{-5}	1.2×10^{-8}	7.5×10^{-9}	2.5×10^{-9}	2.0×10^{-10}	3.0×10^{-10}	4.0×10^{-10}	2.5×10^{-9}

a. The accident is represented by two separate releases.
Key: Am, americium; Ba, barium; Ce, cerium; Cm, curium; Co, cobalt; Cs, cesium; I, iodine; Kr, krypton; La, lanthanum; Mo, molybdenum; Nb, niobium; Nd, neodymium; Np, neptunium; Pu, plutonium; Pr, praseodymium; Rb, rubidium; Rh, rhodium; Ru, ruthenium; Sb, antimony; Sr, strontium; Tc, technetium; Te, tellurium; Xe, xenon; Y, yttrium; Zr, zirconium.
Source: Davis 1997.

NUREG/CR-6295 (Davis 1997) provides frequencies for each accident category. However, these frequencies are based solely on the NUREG-1150 (NRC 1990) plant data. To apply more recent accident frequencies, data from commercial pressurized water reactor IPEs were reviewed. For each of the accident categories (loss-of-coolant accident, early containment failure, late containment failure, and containment bypass) the failure probability medians were calculated. These data represent significant additional risk studies more recent than NUREG-1150.

The frequency of occurrence for the design-basis large-break loss-of-coolant accident is 4.65×10^{-5} per year. This frequency is based on internal initiators (i.e., plant upsets) and does not include external initiators (e.g., earthquakes). External initiators were not included because the frequencies depend solely on site location.

I.1.1.3.6 Beyond-Design-Basis Events

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than reactor design-basis accidents. In reactor design-basis accidents, the mitigating systems are assumed to be available. In severe reactor accidents, even though the initiating event could be a design-basis event (e.g., large-break loss-of-coolant accident), additional failures of mitigating systems would cause some degree of physical deterioration of the fuel in the reactor core and a possible breach of the containment structure leading to the direct release of radioactive materials to the environment.

In NUREG/CR-6295, representative source terms were developed which represent the full spectrum of severe accidents. A small set of source terms was developed by considering release categories which account for a spectrum of possible times and modes of containment failure. For each containment failure mode the source terms were selected based on the dominant accident progression characteristics leading to the containment failure. The magnitudes of releases for each release category were obtained by using the mean values of the probability distributions of source term parameters used in NUREG-1150.

In NUREG/CR-6295, a total of four release categories was selected to represent the spectrum of containment failure modes of the 3,800 megawatts-thermal pressurized water reactor: a containment bypass event, an early containment failure coincident with reactor core vessel breach, a late containment failure, and a no-containment-failure event. The no-containment-failure event is initiated by a large-break loss-of-coolant accident and was used to represent a realistic design-basis large-break loss-of-coolant accident. The containment bypass and failure scenarios are considered beyond-design-basis events and are evaluated in this section.

Containment Bypass. A containment bypass involves failure of the pressure boundary between the high-pressure reactor coolant and low-pressure auxiliary system. For pressurized water reactors, steam generator tube rupture, either as an initiating event or as a result of severe accident conditions, will lead to containment bypass. In these scenarios, if core damage occurs, a direct path to the environment can exist.

Early Containment Failure. This accident is defined as the failure of containment prior to, or very soon (within a few hours) after, breach of the reactor vessel. A variety of mechanisms (e.g., direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, fuel-coolant interactions) can cause structural failure of the containment. Failure to isolate the containment and early containment venting after core damage are also classified as early containment failures.

Late Containment Failure. A late containment failure involves structural failure of the containment several hours after breach of the reactor vessel. A variety of mechanisms (e.g., gradual pressure and temperature increase, hydrogen combustion, basemat melt-through by core debris) can cause late containment failure. Venting the containment late in the accident is also classified as a late containment failure.

The release characteristics for each accident, obtained from NUREG/CR-6295, are described by the release height, timing, duration, and heat content of the plume, the fraction of each isotope group released, and the warning time (time when offsite officials are warned that an emergency response should be initiated). **Tables I–11 and I–12** provide the release parameters for the beyond-design-basis accidents.

Table I–11 Beyond-Design-Basis Accident Release Characteristics

Accident Scenario	Scenario Frequency	Elevation of Release (m)	Energy of Release (W)	Warning Time	Time of Release (hr)	Duration of Release
Containment bypass[a]	1.53×10^{-6}	10	5.5×10^{6}	20 min	1.0	30 min
			9.9×10^{5}		1.5	2 hr
Early containment failure[a]	7.92×10^{-8}	10	8.6×10^{5}	5.0 hr	6.0	10 min
			1.5×10^{6}		6.167	2 hr
Late containment failure	1.07×10^{-5}	10	1.9×10^{5}	5.0 hr	12.0	3 hr

a. The accident is represented by two separate releases.
Key: hr, hour; m, meters; min, minute; W, watts.
Source: Davis 1997.

Table I–12 Beyond-Design-Basis Accident Release Fractions

Accident Scenario	Release Fractions by Isotope								
	Kr, Xe	I	Cs, Rb	Sb, Te	Sr	Co, Mo, Rh, Ru, Tc	Am, Cm, La, Nb, Nd, Pr, Y, Zr	Ce, Np, Pu	Ba
Containment bypass[a]	1.0	7.5×10^{-2}	6.0×10^{-2}	2.0×10^{-2}	5.0×10^{-3}	1.0×10^{-3}	3.0×10^{-4}	1.0×10^{-3}	5.0×10^{-3}
	0.0	4.0×10^{-2}	6.0×10^{-2}	5.0×10^{-2}	2.0×10^{-2}	6.0×10^{-4}	3.0×10^{-3}	3.0×10^{-3}	2.0×10^{-2}
Early containment failure[a]	1.0	2.5×10^{-1}	1.8×10^{-1}	8.0×10^{-2}	2.0×10^{-2}	5.0×10^{-3}	1.0×10^{-3}	5.0×10^{-3}	2.0×10^{-2}
	0.0	2.0×10^{-2}	3.0×10^{-2}	2.0×10^{-2}	1.0×10^{-2}	2.0×10^{-4}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-2}
Late containment failure	1.0	3.0×10^{-2}	6.0×10^{-6}	7.0×10^{-6}	1.0×10^{-6}	2.0×10^{-8}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-6}

a. The accident is represented by two separate releases.
Key: Am, americium; Ba, barium; Ce, cerium; Cm, curium; Co, cobalt; Cs, cesium; I, iodine; Kr, krypton; La, lanthanum; Mo, molybdenum; Nb, niobium; Nd, neodymium; Np, neptunium; Pu, plutonium; Pr, praseodymium; Rb, rubidium; Rh, rhodium; Ru, ruthenium; Sb, antimony; Sr, strontium; Tc, technetium; Te, tellurium; Xe, xenon; Y, yttrium; Zr, zirconium.
Source: Davis 1997.

As in the design-basis-accident analysis, the frequency of occurrence is based on internal initiators and does not include external initiators.

I.1.1.4 Fast Flux Test Facility (FFTF)

A spectrum of postulated accidents was evaluated for three separate FFTF conditions: operation, standby, and deactivation. Conservative assumptions were made on core configuration and isotopic inventory in order to provide conservative estimates of impacts.

I.1.1.4.1 FFTF Operation

For operation, the FFTF core would be modified to include an array of target assemblies and Rapid Radioisotope Retrieval systems to produce cobalt-60, a number of long- and short-lived isotopes for medical applications, and 5 kilograms (11 pounds) per year of plutonium-238 for space power applications. In addition, space is to be provided for research and development test articles such as Accelerator Transmutation of Waste test assemblies.

It is expected that the characteristics of the new mission core will be similar to previous cores, and that the existing facility safety analysis report analyses will be comparable to the new core accidents. A wide range of postulated reactor accidents was analyzed in the existing *FFTF Final Safety Analysis Report* (Dautel 2000).

These include design-basis and beyond-design-basis accidents. A spectrum of postulated accidents was evaluated to provide bounding scenarios for determining potential environmental and health impacts of the new missions. The accident scenarios were selected from the existing *FFTF Final Safety Analysis Report* and represent design-basis and beyond-design-basis events, including reactor, target-handling, and fuel storage accidents. Source terms and frequencies were selected to provide conservative estimates of the potential impacts.

The accident analysis included a review of external events (e.g., nearby explosions, fires), internal events (e.g., equipment failures, human errors), natural phenomena (e.g., floods, earthquakes), as well as sabotage and terrorist acts. A recent external event of concern is the threat of wildfires. Several features of FFTF make it well equipped to deal with an event like a large range fire. First, the layout and construction of the facility make it very unlikely that an external fire would spread to the plant structures (e.g., there is a large clear gravel and asphalt buffer zone, and much of the facility is constructed of fire-resistant materials). Furthermore, most of the critical plant systems, including the reactor and its heat transport system, are housed inside of the steel and concrete containment building, which is completely closed during reactor operation. As appropriate, the balance of the facility is protected by automatic fire detection/suppression systems. Although FFTF has several sources of both offsite and onsite electrical power, another significant safety factor is that, except for a few batteries, FFTF requires no electricity to accomplish any required safety function (i.e., reactor shutdown, isolation of the containment building, and emergency core cooling). Finally, the FFTF control building includes systems such as a filtered air supply for ensuring habitability during a variety of offnormal conditions, and emergency respirators are available to the operators. A wildfire-initiated accident would be bounded by the accidents evaluated and therefore not considered further.

Two large range fires at Hanford burned very close to FFTF. In 1984, a very large fire occurred while the reactor was in operation at 100 percent power. The plant continued to operate normally and safely throughout this event, although a reduction in power was initiated as a precautionary measure. The second fire occurred in June 2000 while the reactor was in standby. In neither case, did the range fire cause any damage or operational difficulties at FFTF. Both fires reached the gravel and asphalt buffer zone around FFTF, but never posed any significant threat to plant structures. Precautionary measures were taken by essential plant personnel to perform continuous monitoring and to reduce or eliminate the intake of smoke passing over the facility.

The reactor power will be 100 megawatts, which is one-fourth of the design power, for most of the mission operation. However, periodic increases in power level between 100 and 400 megawatts may be required to support civilian nuclear energy research and development activities. The accident analyses provided are based on the FFTF design power level of 400 megawatts and will provide conservative estimates of operation at 400 megawatts-thermal and lower power levels.

CORE INVENTORIES

Mixed Oxide Driver Fuel

The current FFTF fuel contains mixed oxide driver fuel assemblies. The plutonium fuel enrichment is assumed to be the same as during previous reactor operations and as currently authorized by the facility safety analysis report. A total of 76 driver fuel assemblies were assumed in the facility safety analysis report. Although it is expected that some of the driver fuel positions will be taken up by test articles and isotope production targets, the same number of driver fuel assemblies are to be assumed for conservatism for purposes of this analysis. A total of six fueled test articles were included in the assumed core loading for this analysis, but were treated as part of the complement of 76 driver fuel assemblies.

An ORIGEN2 (Wootan 1999) calculation for a reference driver fuel assembly was used to generate the radioisotope inventories used in the accident analyses. Evaluation focused on a typical end-of-irradiation inner-row driver fuel assembly with a plutonium enrichment of about 22 percent—specifically, assembly 16439 irradiated to 445.8 effective full-power days through cycle four in core location 1201. Previous studies have determined that 60 isotopes are important for offsite impact analysis. These 60 isotopes are provided in NUREG/CR-4691, *MELCOR Accident Consequence Code System*, Volume 1, Table B.4–2 (Chanin et al. 1990). The resulting driver fuel inventory is shown in **Table I–13**.

Highly Enriched Uranium Driver Fuel

A future core loading may require use of highly enriched uranium. The highly enriched uranium fuel would be in an oxide form. Radioisotope inventories were calculated for a highly enriched uranium fuel assembly that is directly comparable to the reference mixed oxide fuel assembly. To generate comparable values for a highly enriched uranium fueled core, a highly enriched uranium fuel assembly with a uranium-235 enrichment of 25 percent was used to replace the reference mixed oxide assembly in the ORIGEN2 calculation of radioisotope inventories. This enrichment provides about 25 percent more uranium-235 in the highly enriched uranium assembly than plutonium-239 in the mixed oxide assembly, so that the highly enriched uranium assembly would have comparable power and burnup at a lower flux level than the reference mixed oxide assembly. This enrichment is lower than the enrichments expected in a full highly enriched uranium core (likely in the range of 35 percent), but the dose rates for this assembly should bound the higher enrichments, since the fission products would be nearly identical and the plutonium contribution would be less with higher enrichments. The resulting highly enriched uranium driver fuel inventory is shown in **Table I–14**.

Although accidents were evaluated for both the mixed oxide and highly enriched uranium core configurations, it is important to point out that the radiological consequences of the mixed oxide fueled core assumed in this analysis will bound those of the highly enriched uranium core.

Table I–13 FFTF Core Inventory with Mixed Oxide Driver Fuel

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	Core Activity (Ci)
	Driver Activity (Ci)	76 Drivers (6 ATWs)[a] Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[b] (Ci)	12 Ac-227[c] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	
Hydrogen-3	57.56	4,370	0.00241	8.82	–	–	4,380
Cobalt-60	0.000	0.00	0.00	0.00	–	1.48×10^{7}	1.48×10^{7}
Krypton-85	517.6	3.93×10^{4}	0.0202	73.9	–	–	3.94×10^{4}
Krypton-85m	2.850×10^{4}	2.17×10^{6}	5.30	1.94×10^{4}	–	–	2.19×10^{6}
Krypton-87	4.716×10^{4}	3.58×10^{6}	8.83	3.23×10^{4}	–	–	3.62×10^{6}
Krypton-88	6.567×10^{4}	4.99×10^{6}	12.4	4.54×10^{4}	–	–	5.04×10^{6}
Rubidium-86	907.6	6.90×10^{4}	0.00762	27.9	–	–	6.90×10^{4}
Strontium-89	7.598×10^{4}	5.77×10^{6}	9.63	3.52×10^{4}	–	–	5.81×10^{6}
Strontium-90	3,181	2.42×10^{5}	0.127	465	–	–	2.42×10^{5}
Strontium-91	1.205×10^{5}	9.16×10^{6}	23.4	8.56×10^{4}	–	–	9.24×10^{6}
Strontium-92	1.418×10^{5}	1.08×10^{7}	28.4	1.04×10^{5}	–	–	1.09×10^{7}
Yttrium-90	3,561	2.71×10^{5}	0.128	468	–	–	2.71×10^{5}
Yttrium-91	9.984×10^{4}	7.59×10^{6}	13.2	4.83×10^{4}	–	–	7.64×10^{6}
Yttrium-92	1.434×10^{5}	1.09×10^{7}	28.7	1.05×10^{5}	–	–	1.10×10^{7}
Yttrium-93	1.785×10^{5}	1.36×10^{7}	37.2	1.36×10^{5}	–	–	1.37×10^{7}
Zirconium-95	1.776×10^{5}	1.35×10^{7}	24.7	9.04×10^{4}	–	–	1.36×10^{7}
Zirconium-97	2.357×10^{5}	1.79×10^{7}	51.0	1.87×10^{5}	–	–	1.81×10^{7}
Niobium-95	1.492×10^{5}	1.13×10^{7}	16.8	6.15×10^{4}	–	–	1.14×10^{7}
Molybdenum-99	2.690×10^{5}	2.04×10^{7}	56.3	2.06×10^{5}	–	–	2.07×10^{7}
Technetium-99m	2.355×10^{5}	1.79×10^{7}	50.1	1.83×10^{5}	–	–	1.81×10^{7}
Ruthenium-103	2.718×10^{5}	2.07×10^{7}	42.5	1.56×10^{5}	–	–	2.08×10^{7}
Ruthenium-105	2.261×10^{5}	1.72×10^{7}	51.7	1.89×10^{5}	–	–	1.74×10^{7}
Ruthenium-106	9.408×10^{4}	7.15×10^{6}	6.41	2.35×10^{4}	–	–	7.17×10^{6}
Rhodium-105	2.246×10^{5}	1.71×10^{7}	41.1	1.50×10^{5}	–	–	1.72×10^{7}
Antimony-127	2.579×10^{4}	1.96×10^{6}	4.44	1.63×10^{4}	–	–	1.98×10^{6}
Antimony-129	6.280×10^{4}	4.77×10^{6}	13.5	4.94×10^{4}	–	–	4.82×10^{6}
Tellurium-127m	2,535	1.93×10^{5}	0.243	889	–	–	1.94×10^{5}
Tellurium-127	2.471×10^{4}	1.88×10^{6}	4.18	1.53×10^{4}	–	–	1.89×10^{6}
Tellurium-129	6.211×10^{4}	4.72×10^{6}	12.9	4.72×10^{4}	–	–	4.77×10^{6}
Tellurium-129m	8,626	6.56×10^{5}	1.39	5,090	–	–	6.61×10^{5}
Tellurium-131	1.546×10^{5}	1.17×10^{7}	30.9	1.13×10^{5}	–	–	1.19×10^{7}
Tellurium-131m	2.684×10^{4}	2.04×10^{6}	5.96	2.18×10^{4}	–	–	2.06×10^{6}
Tellurium-132	2.314×10^{5}	1.76×10^{7}	47.1	1.72×10^{5}	–	–	1.78×10^{7}
Iodine-125	0.000	0.00	0.00	0.00	2,530	–	2,530
Iodine-131	1.759×10^{5}	1.34×10^{7}	32.5	1.19×10^{5}	–	–	1.35×10^{7}
Iodine-132	2.367×10^{5}	1.80×10^{7}	48.7	1.78×10^{5}	–	–	1.82×10^{7}
Iodine-133	2.996×10^{5}	2.28×10^{7}	65.0	2.38×10^{5}	–	–	2.30×10^{7}
Iodine-134	3.173×10^{5}	2.41×10^{7}	69.0	2.53×10^{5}	–	–	2.44×10^{7}
Iodine-135	2.883×10^{5}	2.19×10^{7}	60.8	2.23×10^{5}	–	–	2.21×10^{7}
Xenon-133	3.051×10^{5}	2.32×10^{7}	61.3	2.24×10^{5}	–	–	2.34×10^{7}
Xenon-135	3.254×10^{5}	2.47×10^{7}	7.69	2.81×10^{4}	–	–	2.48×10^{7}
Cesium-134	4,980	3.78×10^{5}	0.159	582	–	–	3.79×10^{5}

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	Core Activity (Ci)
	Driver Activity (Ci)	76 Drivers (6 ATWs)[a] Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[b] (Ci)	12 Ac-227[c] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	
Cesium-136	9,451	7.18×10^{5}	0.920	3,370	–	–	7.22×10^{5}
Cesium-137	8,361	6.35×10^{5}	0.375	1,370	–	–	6.37×10^{5}
Barium-139	2.594×10^{5}	1.97×10^{7}	54.1	1.98×10^{5}	–	–	1.99×10^{7}
Barium-140	2.397×10^{5}	1.82×10^{7}	45.1	1.65×10^{5}	–	–	1.84×10^{7}
Lanthanum-140	2.421×10^{5}	1.84×10^{7}	44.5	1.63×10^{5}	–	–	1.86×10^{7}
Lanthanum-141	2.460×10^{5}	1.87×10^{7}	51.3	1.88×10^{5}	–	–	1.89×10^{7}
Lanthanum-142	2.179×10^{5}	1.66×10^{7}	47.6	1.74×10^{5}	–	–	1.67×10^{7}
Cerium-141	2.294×10^{5}	1.74×10^{7}	35.0	1.28×10^{5}	–	–	1.76×10^{7}
Cerium-143	1.998×10^{5}	1.52×10^{7}	42.5	1.56×10^{5}	–	–	1.53×10^{7}
Cerium-144	9.360×10^{4}	7.11×10^{6}	7.13	2.61×10^{4}	–	–	7.14×10^{6}
Praseodymium-143	1.966×10^{5}	1.49×10^{7}	35.6	1.30×10^{5}	–	–	1.51×10^{7}
Neodymium-147	9.847×10^{4}	7.48×10^{6}	17.1	6.26×10^{4}	–	–	7.55×10^{6}
Rhenium-186	0.000	0.00	0.00	0.00	3.05×10^{4}	–	3.05×10^{4}
Radium-223	2.644×10^{-9}	2.01×10^{-7}	0.00	0.00	235	–	235
Radium-224	1.245×10^{-4}	0.00946	0.00	0.00	466	–	466
Radium-226	0.000	0.00	0.00	0.00	172	–	172
Actinium-227	1.780×10^{-9}	1.35×10^{-7}	0.00	0.00	408	–	408
Thorium-227	2.714×10^{-9}	2.06×10^{-7}	0.00	0.00	298	–	298
Thorium-228	1.239×10^{-4}	0.00942	0.00	0.00	505	–	505
Neptunium-237	0.009117	0.693	3.60×10^{-3}	13.2	–	–	13.9
Neptunium-239	2.723×10^{6}	2.07×10^{8}	16.8	6.15×10^{4}	–	–	2.07×10^{8}
Plutonium-238	123.6	9,390	17	6.19×10^{4}	–	–	7.12×10^{4}
Plutonium-239	320.1	2.43×10^{4}	9.21×10^{-3}	33.7	–	–	2.44×10^{4}
Plutonium-240	259.3	1.97×10^{4}	3.93×10^{-3}	14.4	–	–	1.97×10^{4}
Plutonium-241	1.213×10^{4}	9.22×10^{5}	0.853	3,120	–	–	9.25×10^{5}
Americium-241	141.1	1.07×10^{4}	0.00	0.00	–	–	1.07×10^{4}
Curium-242	9,829	7.47×10^{5}	0.0122	44.7	–	–	7.47×10^{5}
Curium-244	8.305	631	0.00	0.00	–	–	631

a. Six Accelerator Transmutation of Waste test assemblies included as driver fuel assemblies.
b. Based on a 5-kilogram-per-year plutonium-238 production rate.
c. For the actinium-227 target, over 99.9 percent of the consequences are attributable to six isotopes (actinium-227; radium-223, 224, 226; thorium-227, 228). Therefore, the other actinium-227 target byproducts are not included.

Key: Ac-227, actinium-227; ATW, Accelerator Transmutation of Waste; Ci, curies; Co-60, cobalt-60; I-125, iodine-125; Pu-238, plutonium-238; Re-186, rhenium-186.

Source: BWHC 1999; Schnitzler 1999; Wootan 1999.

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Hydrogen-3	66.04	4,620	0.00241	8.82	–	–	345	4,980
Cobalt-60	0.000	0.00	0.00	0.00	–	1.48×10^{7}	0.00	1.48×10^{7}
Krypton-85	962.1	6.73×10^{4}	0.0202	73.9	–	–	3,110	7.05×10^{4}
Krypton-85m	5.236×10^{4}	3.67×10^{6}	5.30	1.94×10^{4}	–	–	1.71×10^{5}	3.86×10^{6}
Krypton-87	9.952×10^{4}	6.97×10^{6}	8.83	3.23×10^{4}	–	–	2.83×10^{5}	7.28×10^{6}
Krypton-88	1.433×10^{5}	1.00×10^{7}	12.4	4.54×10^{4}	–	–	3.94×10^{5}	1.05×10^{7}
Rubidium-86	1.571×10^{3}	1.10×10^{5}	0.00762	27.9	–	–	5,450	1.15×10^{5}
Strontium-89	1.586×10^{5}	1.11×10^{7}	9.63	3.52×10^{4}	–	–	4.56×10^{5}	1.16×10^{7}
Strontium-90	7.107×10^{3}	4.97×10^{5}	0.127	465	–	–	1.91×10^{4}	5.17×10^{5}
Strontium-91	2.279×10^{5}	1.60×10^{7}	23.4	8.56×10^{4}	–	–	7.23×10^{5}	1.68×10^{7}
Strontium-92	2.385×10^{5}	1.67×10^{7}	28.4	1.04×10^{5}	–	–	8.51×10^{5}	1.76×10^{7}
Yttrium-90	7.885×10^{3}	5.52×10^{5}	0.128	468	–	–	2.14×10^{4}	5.74×10^{5}
Yttrium-91	1.922×10^{5}	1.35×10^{7}	13.2	4.83×10^{4}	–	–	5.99×10^{5}	1.41×10^{7}
Yttrium-92	2.397×10^{5}	1.68×10^{7}	28.7	1.05×10^{5}	–	–	8.60×10^{5}	1.77×10^{7}
Yttrium-93	2.655×10^{5}	1.86×10^{7}	37.2	1.36×10^{5}	–	–	1.07×10^{6}	1.98×10^{7}
Zirconium-95	2.274×10^{5}	1.59×10^{7}	24.7	9.04×10^{4}	–	–	1.07×10^{6}	1.71×10^{7}
Zirconium-97	2.636×10^{5}	1.85×10^{7}	51.0	1.87×10^{5}	–	–	1.41×10^{6}	2.01×10^{7}
Niobium-95	1.921×10^{5}	1.34×10^{7}	16.8	6.15×10^{4}	–	–	8.95×10^{5}	1.44×10^{7}
Molybdenum-99	2.693×10^{5}	1.89×10^{7}	56.3	2.06×10^{5}	–	–	1.61×10^{6}	2.07×10^{7}
Technetium-99m	2.358×10^{5}	1.65×10^{7}	50.1	1.83×10^{5}	–	–	1.41×10^{6}	1.81×10^{7}
Ruthenium-103	1.727×10^{5}	1.21×10^{7}	42.5	1.56×10^{5}	–	–	1.63×10^{6}	1.39×10^{7}
Ruthenium-105	9.789×10^{4}	6.85×10^{6}	51.7	1.89×10^{5}	–	–	1.36×10^{6}	8.40×10^{6}
Ruthenium-106	2.729×10^{4}	1.91×10^{6}	6.41	2.35×10^{4}	–	–	5.82×10^{5}	2.52×10^{6}
Rhodium-105	9.844×10^{4}	6.89×10^{6}	41.1	1.50×10^{5}	–	–	1.35×10^{6}	8.39×10^{6}
Antimony-127	2.172×10^{4}	1.52×10^{6}	4.44	1.63×10^{4}	–	–	1.55×10^{5}	1.69×10^{6}
Antimony-129	5.288×10^{4}	3.70×10^{6}	13.5	4.94×10^{4}	–	–	3.77×10^{5}	4.13×10^{6}
Tellurium-127	2.082×10^{4}	1.46×10^{6}	4.18	1.53×10^{4}	–	–	1.48×10^{5}	1.62×10^{6}

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel (Continued)

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Tellurium-127m	2.134×10^{3}	1.49×10^{5}	0.243	4.72×10^{4}	–	–	3.73×10^{5}	4.18×10^{6}
Tellurium-129	5.366×10^{4}	3.76×10^{6}	12.9	4.72×10^{4}	–	–	3.73×10^{5}	4.18×10^{6}
Tellurium-129m	7,338	5.14×10^{5}	1.39	5,090	–	–	5.18×10^{4}	5.71×10^{5}
Tellurium-131	1.413×10^{5}	9.89×10^{6}	30.9	1.13×10^{5}	–	–	9.28×10^{5}	1.09×10^{7}
Tellurium-131m	2.028×10^{4}	1.42×10^{6}	5.96	2.18×10^{4}	–	–	1.61×10^{5}	1.60×10^{6}
Tellurium-132	2.174×10^{5}	1.52×10^{7}	47.1	1.72×10^{5}	–	–	1.39×10^{6}	1.68×10^{7}
Iodine-125	0.000	0.00	0.00	0.00	2,530	–	0.00	2,530
Iodine-131	1.574×10^{5}	1.10×10^{7}	32.5	1.19×10^{5}	–	–	1.06×10^{6}	1.22×10^{7}
Iodine-132	2.209×10^{5}	1.55×10^{7}	48.7	1.78×10^{5}	–	–	1.42×10^{6}	1.71×10^{7}
Iodine-133	2.996×10^{5}	2.10×10^{7}	65.0	2.38×10^{5}	–	–	1.80×10^{6}	2.30×10^{7}
Iodine-134	3.412×10^{5}	2.39×10^{7}	69.0	2.53×10^{5}	–	–	1.90×10^{6}	2.60×10^{7}
Iodine-135	2.763×10^{5}	1.93×10^{7}	60.8	2.23×10^{5}	–	–	1.73×10^{6}	2.13×10^{7}
Xenon-133	3.034×10^{5}	2.12×10^{7}	61.3	2.24×10^{5}	–	–	1.83×10^{6}	2.33×10^{7}
Xenon-135	2.995×10^{5}	2.10×10^{7}	7.69	2.81×10^{4}	–	–	1.95×10^{6}	2.29×10^{7}
Cesium-134	4,676	3.27×10^{5}	0.159	582	–	–	2.99×10^{4}	3.58×10^{5}
Cesium-136	5,314	3.72×10^{5}	0.920	3,370	–	–	5.67×10^{4}	4.32×10^{5}
Cesium-137	8,319	5.82×10^{5}	0.375	1,370	–	–	5.02×10^{4}	6.34×10^{5}
Barium-139	2.833×10^{5}	1.98×10^{7}	54.1	1.98×10^{5}	–	–	1.56×10^{6}	2.16×10^{7}
Barium-140	2.674×10^{5}	1.87×10^{7}	45.1	1.65×10^{5}	–	–	1.44×10^{6}	2.03×10^{7}
Lanthanum-140	2.698×10^{5}	1.89×10^{7}	44.5	1.63×10^{5}	–	–	1.45×10^{6}	2.05×10^{7}
Lanthanum-141	2.658×10^{5}	1.86×10^{7}	51.3	1.88×10^{5}	–	–	1.48×10^{6}	2.03×10^{7}
Lanthanum-142	2.461×10^{5}	1.72×10^{7}	47.6	1.74×10^{5}	–	–	1.31×10^{6}	1.87×10^{7}
Cerium-141	2.492×10^{5}	1.74×10^{7}	35.0	1.28×10^{5}	–	–	1.38×10^{6}	1.89×10^{7}
Cerium-143	2.467×10^{5}	1.73×10^{7}	42.5	1.56×10^{5}	–	–	1.20×10^{6}	1.86×10^{7}
Cerium-144	1.277×10^{5}	8.94×10^{6}	7.13	2.61×10^{4}	–	–	5.62×10^{5}	9.53×10^{6}
Praseodymium-143	2.437×10^{5}	1.71×10^{7}	35.6	1.30×10^{5}	–	–	1.18×10^{6}	1.84×10^{7}

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel (Continued)

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Neodymium-147	1.098×10^{5}	7.69×10^{6}	17.1	6.26×10^{4}	–	–	5.91×10^{5}	8.34×10^{6}
Rhenium-186	0.000	0.00	0.00	0.00	3.05×10^{4}	–	0.00	3.64×10^{4}
Radium-223	8.279×10^{-8}	5.80×10^{-6}	0.00	0.00	235	–	1.59×10^{-8}	235
Radium-224	2.260×10^{-4}	0.0158	0.00	0.00	466	–	7.47×10^{-4}	466
Radium-226	3.80×10^{-11}	2.66×10^{-9}	0.00	0.00	172	–	0.00	172
Actinium-227	8.421×10^{-8}	5.89×10^{-6}	0.00	0.00	408	–	1.07×10^{-8}	408
Thorium-227	8.293×10^{-8}	5.81×10^{-6}	0.00	0.00	298	–	1.63×10^{-8}	298
Thorium-228	2.251×10^{-4}	0.0158	0.00	0.00	505	–	7.43×10^{-4}	505
Neptunium-237	0.02577	1.80	0.00360	13.2	–	–	0.0547	15.0
Neptunium-239	2.406×10^{6}	1.68×10^{8}	16.8	6.15×10^{4}	–	–	1.63×10^{7}	1.85×10^{8}
Plutonium-238	57.38	4,020	17	6.19×10^{4}	–	–	742	6.66×10^{4}
Plutonium-239	68.66	4,810	0.00921	33.7	–	–	1,920	6,760
Plutonium-240	10.45	732	0.00393	14.4	–	–	1,560	2,300
Plutonium-241	132.2	9,250	0.853	3,120	–	–	7.28×10^{4}	8.52×10^{4}
Americium-241	0.08854	6.20	0.00	0.00	–	–	847	853
Curium-242	2.844	199	0.0122	44.7	–	–	6.06×10^{4}	6.08×10^{4}
Curium-244	9.215×10^{-4}	0.0645	0.00	0.00	–	–	51.2	51.3

a. Based on a 5-kilogram-per-year plutonium-238 production rate.
b. For the actinium-227 target, over 99.9 percent of the consequences are attributable to six isotopes (actinium-227; radium-223, 224, 226; thorium-227, 228). Therefore, the other actinium-227 target byproducts are not included.
c. Six Accelerator Transmutation of Waste test assemblies included as mixed oxide driver fuel assemblies.

Key: Ac-227, actinium-227; ATW, Accelerator Transmutation of Waste; Ci, curies; Co-60, cobalt-60; HEU, highly enriched uranium; I-125, iodine-125; Pu-238, plutonium-238; Re-186, rhenium-186.

Source: BWHC 1999; Schnitzler 1999; Wootan 1999.

Targets

The proposed core modifications include an array of target assemblies and Rapid Radioisotope Retrieval systems to produce plutonium-238 for space power applications, cobalt-60, and a number of long- and short-lived isotopes for medical applications. In addition, space is to be provided for research and development test articles such as Accelerator Transmutation of Waste test assemblies. As stated previously, a total of six Accelerator Transmutation of Waste test assemblies were conservatively modeled as mixed oxide driver fuel assemblies and included as part of the complement of 76 driver fuel assemblies.

To determine which Rapid Radioisotope Retrieval system and Long-Term Irradiation Vehicle irradiated targets would result in the maximum consequences, the radioisotope inventories for each of the irradiated targets were multiplied by the same release fractions as were assumed for the fuel and fission products (1 percent for solids and 100 percent for noble gases). The resulting inventories were then multiplied by dose conversion factors resulting in a dose for each isotope. The isotope doses within each target were totaled for a target dose, and the target doses were compared to determine which target would result in the maximum consequence for each target type.

Rapid Radioisotope Retrieval Systems

There is to be a maximum of eight Rapid Radioisotope Retrieval systems in the core. One of the Rapid Radioisotope Retrieval systems is to be configured as a gas target to produce iodine-125 from xenon-124. The other seven will be used for production of solid, short-lived medical isotopes. These seven targets are all modeled as the worst-case type (other than gas) to maximize the resulting dose contribution of an accident. The worst-case target planned for insertion in a Rapid Radioisotope Retrieval system is the xenon-124 gas tube, which is assumed to release 100 percent of its iodine-125 inventory along with the xenon-124 gas into containment in the event of any break in the system. As the next worst is the rhenium-186 production target, the other seven Rapid Radioisotope Retrieval systems were assumed to be rhenium-186 production targets. The Rapid Radioisotope Retrieval system inventory is shown in Tables I–13 and I–14.

Long-Term Irradiation Vehicle

Twelve Long-Term Irradiation Vehicle assemblies for production of long-lived medical isotopes are assumed. These assemblies are all modeled as the worst-case type to maximize the dose contribution of an accident. The worst-case Long-Term Irradiation Vehicle target is the actinium-227 production target. All 12 Long-Term Irradiation Vehicle targets are therefore assumed to be actinium-227 production targets. The Long-Term Irradiation Vehicle inventory is shown in Tables I–13 and I–14.

Cobalt-60 Production Target Assemblies

Forty-eight cobalt-60 production targets are to be included in row 9 (outside the reflector assemblies) with a currently assumed annual production rate of 2.016×10^7 curies. The residence time for these targets is to be three 100-day cycles with 16 assemblies being harvested at the end of each cycle. Assuming 2.73 cycles per year for FFTF, this leads to a maximum end-of-cycle core inventory of $(1/3 + 2/3 + 1)$ $(2.016\times10^7 \text{ curies})/2.73 = 1.48\times10^7$ curies. This inventory is included in Tables I–13 and I–14.

Plutonium-238 Production Target Assemblies

Fifteen plutonium-238 production targets are to be included in the reflector region with a currently assumed annual production rate of 5 kilograms (11 pounds). The residence time for these targets is to be three 100-day cycles with five assemblies being harvested at the end of each cycle. Assuming 2.73 cycles per year for FFTF,

this leads to a maximum end-of-cycle core inventory of (1/3 + 2/3 + 1) (5 kilograms)/2.73 = 3.66 kilograms (8.07 pounds). The end-of-cycle target inventory per gram of plutonium-238 and the associated maximum end-of-cycle inventory (3.66 kilograms [8.07 pounds]) are shown in Tables I–13 and I–14.

DESIGN-BASIS ACCIDENT

A wide range of design-basis accidents is analyzed in Chapter 15 of the *FFTF Final Safety Analysis Report* (Dautel 2000). The reactor accidents include various reactivity increase and heat removal reduction transients as well as local fuel failure and natural phenomena (e.g., seismic) events. It should be noted that the *FFTF Final Safety Analysis Report* does not specifically identify a probability of occurrence for each event, but does classify each as being in the anticipated, unlikely, or extremely unlikely category.

For the design-basis transients, the reactor shutdown system was shown to initiate automatic reactor shutdown (scram) in sufficient time to maintain calculated cladding temperatures/strains within limits that ensured that the integrity of the fuel cladding was maintained. Postulated local fuel failure events were shown to remain local (not propagate) and thus have minimal radiological consequences. The core characteristics (physical, nuclear, and thermal) used in the accident analyses of the final safety analysis report were selected to bound those for any anticipated core design. Also, the characteristics of the new-missions core are expected to be similar to those of previous cores (at the same power level). Therefore, the design-basis transients for the new missions core are expected to be essentially the same as those documented in the final safety analysis report. However, the isotopic inventory of the new-missions core will be different from that is used in the final safety analysis report.

The accidental release of primary sodium from the main heat transport system resulted in a larger radiological release than any other reactor related design-basis event (radiological releases can also occur due to non-reactor-related events, such as fuel-handling accidents). The analyses given in the final safety analysis report include some large spills of primary sodium but the spills involving primary sodium are in cells which are inerted and located within containment. The amount of radioactivity released to the environment is bounded by the main heat transport system spill. Secondary-loop sodium does not contain significant radioactive materials, so the radiological consequences of secondary sodium spills are negligible.

Primary sodium is radioactive and may also contain small amounts of fission products. Sodium temperatures are maintained at less than 566 °C (1,050 °F), much below the sodium boiling point (881 °C [1,618 °F] at atmospheric pressure). Sodium at this temperature will retain practically all of the fission products dissolved in it (except the noble gases). Therefore, sodium itself provides the first barrier to the release of any radioactive species.

Primary sodium is contained in high-integrity stainless steel piping and vessels, which provide the second barrier to release. An additional safety margin is provided by the low system operating pressure (less than 200 pounds per square inch gage). The primary sodium systems are located in inert-gas-filled cells (nitrogen plus 0.8–1.2 volume-percent oxygen) to preclude sustained burning in the event of a spill. Therefore, the cell temperature and pressure rise due to primary sodium spills are minimized. An additional protective feature is the sensitive primary sodium leak detection system, including detectors which annunciate on low oxygen level in the cell atmosphere. These cells are constructed of reinforced concrete several feet thick and completely lined with welded steel plate. They provide the third barrier to radioactivity release in case of a primary sodium spill. These subgrade primary cells are located within the Containment Building, the fourth barrier to the release of radioactivity. Some low-pressure, low inventory auxiliary sodium systems, which are connected to the main primary sodium coolant system, are located in similar concrete, steel-lined subgrade cells in a building adjacent to the Containment Building.

The limiting accident is a spill of primary sodium in the inerted sodium and argon sampling pipeway located outside of containment in the Heat Transport System Building–South. The spill is assumed to occur with the reactor at full power (400 megawatts) and very conservatively assumed to be operating with 1 percent failed fuel. In other words, it is assumed that 1 percent of the radioactive inventory (fission products and actinides plus daughters) of 76 fuel assemblies is dispersed uniformly in the primary sodium. Radioactive gases would not be dispersed in the sodium and therefore would not be available for release in this accident. However, for conservatism, 1 percent of the radioactive gas inventory was included in the analysis.

The mass of primary sodium in the main heat transport system is 421,940 kilograms (930,220 pounds). In addition to fission products and fuel, sodium activation products (sodium-22 and sodium-24) will be created during irradiation. The equilibrium sodium activity is assumed to be 4.1×10^{-4} curie per pound for sodium-22 and 5.38 curies per pound for sodium-24. The final safety analysis report assumes that 393 kilograms (867 pounds) of primary sodium is spilled in the inerted sodium and argon sampling pipeway and that 24.4 kilograms (53.7 pounds) of the total spill burns to form an airborne oxide. The fraction of radioactive inventory (sodium activation, fission products and actinides plus daughters) available for release from the sodium and argon sampling pipeway is $(0.01 \times 53.7)/930{,}220 = 5.77\times10^{-7}$. The leak rate of the sodium and argon sampling pipeway is assumed to be 25 percent per day. The probability of this event is judged to be extremely unlikely (1×10^{-4} to 1×10^{-6} per year). For this NI PEIS, the probability is conservatively chosen to be 1×10^{-4} per year.

The source terms for the design-basis sodium spill with mixed oxide fuel and highly enriched uranium fuel are presented in **Tables I–15 and I–16**, respectively.

It should be noted that the reactor power will be 100 megawatts or one-fourth of the design power, for most of the new-missions operation. The fission production rate will be less for this lower-power operation. Therefore, the actual inventory of radioisotopes will likely be less than the conservative bounding inventory assumed for this analysis.

Table I–15 Design-Basis-Accident Source Term—Mixed Oxide Fuel

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,380	5.77×10^{-7}	0.00253
Sodium-22	381	5.77×10^{-5}	0.0220
Sodium-24	5.00×10^{6}	5.77×10^{-5}	289
Cobalt-60	1.48×10^{7}	5.77×10^{-7}	8.54
Krypton-85	3.94×10^{4}	5.77×10^{-7}	0.0227
Krypton-85m	2.19×10^{6}	5.77×10^{-7}	1.26
Krypton-87	3.62×10^{6}	5.77×10^{-7}	2.09
Krypton-88	5.04×10^{6}	5.77×10^{-7}	2.91
Rubidium-86	6.90×10^{4}	5.77×10^{-7}	0.0398
Strontium-89	5.81×10^{6}	5.77×10^{-7}	3.35
Strontium-90	2.42×10^{5}	5.77×10^{-7}	0.140
Strontium-91	9.24×10^{6}	5.77×10^{-7}	5.33
Strontium-92	1.09×10^{7}	5.77×10^{-7}	6.28
Yttrium-90	2.71×10^{5}	5.77×10^{-7}	0.156
Yttrium-91	7.64×10^{6}	5.77×10^{-7}	4.41
Yttrium-92	1.10×10^{7}	5.77×10^{-7}	6.35
Yttrium-93	1.37×10^{7}	5.77×10^{-7}	7.91
Zirconium-95	1.36×10^{7}	5.77×10^{-7}	7.84
Zirconium-97	1.81×10^{7}	5.77×10^{-7}	10.4
Niobium-95	1.14×10^{7}	5.77×10^{-7}	6.58
Molybdenum-99	2.07×10^{7}	5.77×10^{-7}	11.9
Technetium-99m	1.81×10^{7}	5.77×10^{-7}	10.4
Ruthenium-103	2.08×10^{7}	5.77×10^{-7}	12.0
Ruthenium-105	1.74×10^{7}	5.77×10^{-7}	10.0
Ruthenium-106	7.17×10^{6}	5.77×10^{-7}	4.14
Rhodium-105	1.72×10^{7}	5.77×10^{-7}	9.94
Antimony-127	1.98×10^{6}	5.77×10^{-7}	1.14
Antimony-129	4.82×10^{6}	5.77×10^{-7}	2.78
Iodine-125	2,530	5.77×10^{-7}	0.00146
Tellurium-127	1.89×10^{6}	5.77×10^{-7}	1.09
Tellurium-127m	1.94×10^{5}	5.77×10^{-7}	0.112
Tellurium-129	4.77×10^{6}	5.77×10^{-7}	2.75
Tellurium-129m	6.61×10^{5}	5.77×10^{-7}	0.381
Tellurium-131	1.19×10^{7}	5.77×10^{-7}	6.84
Tellurium-131m	2.06×10^{6}	5.77×10^{-7}	1.19
Tellurium-132	1.78×10^{7}	5.77×10^{-7}	10.2
Iodine-131	1.35×10^{7}	5.77×10^{-7}	7.78
Iodine-132	1.82×10^{7}	5.77×10^{-7}	10.5
Iodine-133	2.30×10^{7}	5.77×10^{-7}	13.3
Iodine-134	2.44×10^{7}	5.77×10^{-7}	14.1
Iodine-135	2.21×10^{7}	5.77×10^{-7}	12.8
Xenon-133	2.34×10^{7}	5.77×10^{-7}	13.5
Xenon-135	2.48×10^{7}	5.77×10^{-7}	14.3
Cesium-134	3.79×10^{5}	5.77×10^{-7}	0.219
Cesium-136	7.22×10^{5}	5.77×10^{-7}	0.416
Cesium-137	6.37×10^{5}	5.77×10^{-7}	0.367
Barium-139	1.99×10^{7}	5.77×10^{-7}	11.5
Barium-140	1.84×10^{7}	5.77×10^{-7}	10.6
Lanthanum-140	1.86×10^{7}	5.77×10^{-7}	10.7

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-141	1.89×10^{7}	5.77×10^{-7}	10.9
Lanthanum-142	1.67×10^{7}	5.77×10^{-7}	9.66
Cerium-141	1.76×10^{7}	5.77×10^{-7}	10.1
Cerium-143	1.53×10^{7}	5.77×10^{-7}	8.85
Cerium-144	7.14×10^{6}	5.77×10^{-7}	4.12
Praseodymium-143	1.51×10^{7}	5.77×10^{-7}	8.70
Neodymium-147	7.55×10^{6}	5.77×10^{-7}	4.35
Rhenium-186	3.64×10^{4}	5.77×10^{-7}	0.0210
Radium-223	235	5.77×10^{-7}	1.36×10^{-4}
Radium-224	466	5.77×10^{-7}	2.69×10^{-4}
Radium-226	172	5.77×10^{-7}	9.92×10^{-5}
Actinium-227	408	5.77×10^{-7}	2.35×10^{-4}
Thorium-227	298	5.77×10^{-7}	1.72×10^{-4}
Thorium-228	505	5.77×10^{-7}	2.91×10^{-4}
Neptunium-237	13.9	5.77×10^{-7}	8.00×10^{-6}
Neptunium-239	2.07×10^{8}	5.77×10^{-7}	119
Plutonium-238	7.12×10^{4}	5.77×10^{-7}	0.0411
Plutonium-239	2.44×10^{4}	5.77×10^{-7}	0.0141
Plutonium-240	1.97×10^{4}	5.77×10^{-7}	0.0114
Plutonium-241	9.25×10^{5}	5.77×10^{-7}	0.534
Americium-241	1.07×10^{4}	5.77×10^{-7}	0.00619
Curium-242	7.47×10^{5}	5.77×10^{-7}	0.431
Curium-244	631	5.77×10^{-7}	3.64×10^{-4}

Source: Nielsen 1999.

Table I–16 Design-Basis-Accident Source Term—Highly Enriched Uranium Fuel

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,980	5.77×10^{-7}	0.00287
Sodium-22	381	5.77×10^{-5}	0.0220
Sodium-24	5.00×10^{6}	5.77×10^{-5}	289
Cobalt-60	1.48×10^{7}	5.77×10^{-7}	8.54
Krypton-85	7.05×10^{4}	5.77×10^{-7}	0.0407
Krypton-85m	3.86×10^{6}	5.77×10^{-7}	2.22
Rubidium-86	1.15×10^{5}	5.77×10^{-7}	0.0666
Krypton-87	7.28×10^{6}	5.77×10^{-7}	4.20
Krypton-88	1.05×10^{7}	5.77×10^{-7}	6.04
Strontium-89	1.16×10^{7}	5.77×10^{-7}	6.69
Strontium-90	5.17×10^{5}	5.77×10^{-7}	0.298
Strontium-91	1.68×10^{7}	5.77×10^{-7}	9.67
Strontium-92	1.76×10^{7}	5.77×10^{-7}	10.2
Yttrium-90	5.74×10^{5}	5.77×10^{-7}	0.331
Yttrium-91	1.41×10^{7}	5.77×10^{-7}	8.14
Yttrium-92	1.77×10^{7}	5.77×10^{-7}	10.2
Yttrium-93	1.98×10^{7}	5.77×10^{-7}	11.4
Zirconium-95	1.71×10^{7}	5.77×10^{-7}	9.85
Zirconium-97	2.01×10^{7}	5.77×10^{-7}	11.6
Niobium-95	1.44×10^{7}	5.77×10^{-7}	8.31
Molybdenum-99	2.07×10^{7}	5.77×10^{-7}	11.9
Technetium-99m	1.81×10^{7}	5.77×10^{-7}	10.4
Ruthenium-103	1.39×10^{7}	5.77×10^{-7}	8.01
Ruthenium-105	8.40×10^{6}	5.77×10^{-7}	4.85
Ruthenium-106	2.52×10^{6}	5.77×10^{-7}	1.45
Rhodium-105	8.39×10^{6}	5.77×10^{-7}	4.84
Antimony-127	1.69×10^{6}	5.77×10^{-7}	0.976
Antimony-129	4.13×10^{6}	5.77×10^{-7}	2.38
Tellurium-127m	1.65×10^{5}	5.77×10^{-7}	0.0955
Tellurium-127	1.62×10^{6}	5.77×10^{-7}	0.935
Tellurium-129	4.18×10^{6}	5.77×10^{-7}	2.41
Tellurium-129m	5.71×10^{5}	5.77×10^{-7}	0.329
Tellurium-131	1.09×10^{7}	5.77×10^{-7}	6.31
Tellurium-131m	1.60×10^{6}	5.77×10^{-7}	0.925
Tellurium-132	1.68×10^{7}	5.77×10^{-7}	9.68
Iodine-125	2,530	5.77×10^{-7}	0.00146
Iodine-131	1.22×10^{7}	5.77×10^{-7}	7.03
Iodine-132	1.71×10^{7}	5.77×10^{-7}	9.84
Iodine-133	2.30×10^{7}	5.77×10^{-7}	13.3
Iodine-134	2.60×10^{7}	5.77×10^{-7}	15.0
Iodine-135	2.13×10^{7}	5.77×10^{-7}	12.3
Xenon-133	2.33×10^{7}	5.77×10^{-7}	13.4
Xenon-135	2.29×10^{7}	5.77×10^{-7}	13.2
Cesium-134	3.58×10^{5}	5.77×10^{-7}	0.206
Cesium-136	4.32×10^{5}	5.77×10^{-7}	0.249
Cesium-137	6.34×10^{5}	5.77×10^{-7}	0.366
Barium-139	2.16×10^{7}	5.77×10^{-7}	12.5
Barium-140	2.03×10^{7}	5.77×10^{-7}	11.7

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-140	2.05×10^{7}	5.77×10^{-7}	11.8
Lanthanum-141	2.03×10^{7}	5.77×10^{-7}	11.7
Lanthanum-142	1.87×10^{7}	5.77×10^{-7}	10.8
Cerium-141	1.89×10^{7}	5.77×10^{-7}	10.9
Cerium-143	1.86×10^{7}	5.77×10^{-7}	10.7
Cerium-144	9.53×10^{6}	5.77×10^{-7}	5.50
Praseodymium-143	1.84×10^{7}	5.77×10^{-7}	10.6
Neodymium-147	8.34×10^{6}	5.77×10^{-7}	4.81
Rhenium-186	3.64×10^{4}	5.77×10^{-7}	0.0210
Radium-223	235	5.77×10^{-7}	1.36×10^{-4}
Radium-224	466	5.77×10^{-7}	2.69×10^{-4}
Radium-226	172	5.77×10^{-7}	9.92×10^{-5}
Actinium-227	408	5.77×10^{-7}	2.35×10^{-4}
Thorium-227	298	5.77×10^{-7}	1.72×10^{-4}
Thorium-228	505	5.77×10^{-7}	2.91×10^{-4}
Neptunium-237	15.0	5.77×10^{-7}	8.67×10^{-6}
Neptunium-239	1.85×10^{8}	5.77×10^{-7}	107
Plutonium-238	6.66×10^{4}	5.77×10^{-7}	0.0384
Plutonium-239	6,760	5.77×10^{-7}	0.00390
Plutonium-240	2,300	5.77×10^{-7}	0.00133
Plutonium-241	8.52×10^{4}	5.77×10^{-7}	0.0491
Americium-241	853	5.77×10^{-7}	4.92×10^{-4}
Curium-242	6.08×10^{4}	5.77×10^{-7}	0.0351
Curium-244	51.3	5.77×10^{-7}	2.96×10^{-5}

Source: Nielsen 1999.

SEVERE REACTOR ACCIDENT

In addition to the design-basis accidents analyzed in Chapter 15 of the *FFTF Final Safety Analysis Report* (Dautel 2000), Appendix A of the facility safety analysis report documents the analysis of two beyond-design-basis events: unprotected transient overpower and unprotected loss of primary sodium flow (unprotected refers to the assumption that the reactor shutdown system fails to shut down the reactor). These two unprotected events are considered to bound the consequences of other potential beyond-design-basis events such as loss of decay heat removal capability.

The unprotected transient overpower event was found to be relatively benign (i.e., no substantial release of radioactive material is expected). The final safety analysis report results indicated that the event would be terminated by fuel melting and sweepout from a few fuel assemblies and in-place cooling of the remainder of the core. There was no identified source of substantial energetics that would challenge the integrity of the reactor vessel, primary heat transport system, or containment boundaries.

In the case of the unprotected loss-of-flow event, meltdown of the entire core could not be precluded, and release of the entire core contents to the primary heat transport system could occur. Extensive analysis showed, however, that a core meltdown does not threaten the integrity of the reactor vessel or primary heat transport system. The core contents are released and severely contaminate the primary system, but are not expected to leak from the primary boundary. Although a relatively benign scenario of fuel melting/boilout was predicted, the possibility of energetics from either large-reactivity insertion events or hot-core interaction with outlet plenum sodium (rapid generation/expansion of sodium vapor) could not be precluded. Conservative estimates of the energy releases from these scenarios were made, and it was shown that the reactor vessel, primary heat

transport system, and containment boundaries would remain intact (although some primary sodium was calculated to be expelled through reactor head seals into the Containment Building due to sodium slug impact on the underside of the reactor head).

Since the neutronic and thermal-hydraulic conditions for the proposed new-missions core are expected to be similar to those for the previous FFTF cores (at 400 megawatts), the severe-accident scenarios are also expected to be similar. It should be noted that lower-power (100-megawatt) operation would reduce the severity of severe accidents. It is assumed that all the core fuel assemblies and reflector target assemblies eventually melt during the unprotected loss-of-flow accident. It is further assumed that no mitigating actions are taken to restore core cooling during the event, and that, as a bounding case, all fuel assemblies melt immediately after reactor shutdown with no decay time prior to release from containment, and that an energetic sodium release into containment occurs consistent with the final safety analysis report–stipulated unprotected loss-of-flow accident.

In the final safety analysis report analysis, 136 kilograms (300 pounds) of sodium was assumed to spray into the containment and burn, thereby heating and pressurizing the containment atmosphere. This provides the driving force for leakage from the containment into the environment. The inclusion of up to eight Rapid Radioisotope Retrieval systems provides additional potential leakage paths for sodium ejection into the Containment Building during sodium slug impact on the bottom of the reactor head. An increase in the quantity of sodium ejected from the primary system would cause increased leakage into the environment.

According to the current conceptual design for the Rapid Radioisotope Retrieval system, each system uses a target tube with an inside diameter of 1.89 centimeters (0.745 inch). This leads to a leakage area for eight systems (including the gas target) of 22.5 square centimeters (3.49 square inches). Assuming that the leak rate is proportional to the leakage area leads to an estimated total leakage of approximately 336 kilograms (740 pounds) of sodium from the eight Rapid Radioisotope Retrieval positions during a postulated unprotected loss-of-flow accident. To account for uncertainty in the calculation and to add conservatism to the evaluation, a leakage of 363 kilograms (800 pounds) of sodium was specified as the contribution from the Rapid Radioisotope Retrieval systems. This increases the total sodium leakage into containment from 136 kilograms (300 pounds) to 499 kilograms (1,100 pounds). In addition to the sodium, 100 percent of the noble gases and one percent of the core fuel and fission product inventory were assumed to be released to the containment.

One of the effects of an additional amount of sodium being sprayed into the containment and burned is to increase the pressurization of the containment and hence the amount and rate of release from the containment. Heating of the containment atmosphere due to the 136-kilograms (300-pound) sodium spill resulted in a peak containment pressure of 1.84 pounds per square inch gage. The revised analysis assumes that 499 kilograms (1,100 pounds) of sodium is ejected into the containment, increasing the peak containment pressure to 4.99 pounds per square inch gage.

The sodium is assumed to mix uniformly with the air in the containment and burn completely. In addition, all the heat conducted from the sodium due to its elevated temperature and all of the heat of combustion are used to heat the air in the containment, with no transmission to the walls or structure. The resulting peak containment pressure of 4.99 pounds per square inch gage is well below the containment design pressure of 10 pounds per square inch gage. The release from the containment building is based on the design release rate of 0.5 percent per day for the duration of the pressure buildup (approximately 225 hours).

Assuming 100 percent of the noble gases and tritium in the core is released to the containment, and the containment leaks at 0.5 percent per day for 24 hours, the release fraction for noble gases and tritium is 0.05 ($.005/24 \times 225 = 0.047 \approx 0.05$). Assuming 1 percent of the fission products, fuel, and target inventory is released to the containment, the release fraction for these isotopes is 5×10^{-4} (0.05×0.01). Assuming

499 kilograms (1,100 pounds) of the 421,940 kilograms (930,220 pounds) of primary sodium is released to the containment, the release fraction for sodium is 5.92×10^{-5} (0.05 × 1,100/930,220).

The unprotected loss-of-flow event, resulting in a complete core melt, represents the most severe accident analyzed for FFTF. The frequency of this event was estimated to be 1×10^{-9} per year, based on internally initiated events (Dautel 2000). For this analysis, the frequency was increased to 1×10^{-6} to incorporate the spectrum of externally initiated events that could contribute to the severe core melt scenario. The main contributor to the increased frequency is a catastrophic earthquake. The magnitude of potential earthquakes with return periods greater than 10,000 years is highly uncertain. For the purposes of this NI PEIS, it was assumed that an earthquake with a return period of 1 million years would result in sufficient ground motion to cause major damage to FFTF resulting in a core melt scenario. An earthquake of this magnitude could result in severe effects to the entire region, including building collapses, power outages, and road hazards.

The source terms for the beyond-design-basis core melt accident with mixed oxide fuel and highly enriched uranium fuel are presented in **Tables I–17** and **I–18**, respectively.

Table I–17 Beyond-Design-Basis Accident Source Term—Mixed Oxide Fuel

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,380	0.05	219
Sodium-22	381	5.92×10^{-5}	0.0226
Sodium-24	5.00×10^{6}	5.92×10^{-5}	296
Cobalt-60	1.48×10^{7}	5×10^{-4}	7,400
Krypton-85	3.94×10^{4}	0.05	1,971
Krypton-85m	2.19×10^{6}	0.05	1.09×10^{5}
Krypton-87	3.62×10^{6}	0.05	1.81×10^{5}
Krypton-88	5.04×10^{6}	0.05	2.52×10^{5}
Rubidium-86	6.90×10^{4}	5×10^{-4}	34.5
Strontium-89	5.81×10^{6}	5×10^{-4}	2,905
Strontium-90	2.42×10^{5}	5×10^{-4}	121
Strontium-91	9.24×10^{6}	5×10^{-4}	4,622
Strontium-92	1.09×10^{7}	5×10^{-4}	5,440
Yttrium-90	2.71×10^{5}	5×10^{-4}	136
Yttrium-91	7.64×10^{6}	5×10^{-4}	3,818
Yttrium-92	1.10×10^{7}	5×10^{-4}	5,502
Yttrium-93	1.37×10^{7}	5×10^{-4}	6,851
Zirconium-95	1.36×10^{7}	5×10^{-4}	6,794
Zirconium-97	1.81×10^{7}	5×10^{-4}	9,050
Niobium-95	1.14×10^{7}	5×10^{-4}	5,700
Molybdenum-99	2.07×10^{7}	5×10^{-4}	1.03×10^{4}
Technetium-99m	1.81×10^{7}	5×10^{-4}	9,041
Ruthenium-103	2.08×10^{7}	5×10^{-4}	1.04×10^{4}
Ruthenium-105	1.74×10^{7}	5×10^{-4}	8,686
Ruthenium-106	7.17×10^{6}	5×10^{-4}	3,698
Rhodium-105	1.72×10^{7}	5×10^{-4}	8,610
Antimony-127	1.98×10^{6}	5×10^{-4}	988
Antimony-129	4.82×10^{6}	5×10^{-4}	2,411
Tellurium-127	1.89×10^{6}	5×10^{-4}	947
Tellurium-127m	1.94×10^{5}	5×10^{-4}	96.8
Tellurium-129	4.77×10^{6}	5×10^{-4}	2,383
Tellurium-129m	6.61×10^{5}	5×10^{-4}	330
Tellurium-131	1.19×10^{7}	5×10^{-4}	5,931
Tellurium-131m	2.06×10^{6}	5×10^{-4}	1,031
Tellurium-132	1.78×10^{7}	5×10^{-4}	8,879
Iodine-125	2,530	5×10^{-4}	1.30
Iodine-131	1.35×10^{7}	5×10^{-4}	6,744
Iodine-132	1.82×10^{7}	5×10^{-4}	9,084
Iodine-133	2.30×10^{7}	5×10^{-4}	1.15×10^{4}
Iodine-134	2.44×10^{7}	5×10^{-4}	1.22×10^{4}
Iodine-135	2.21×10^{7}	5×10^{-4}	1.11×10^{4}
Xenon-133	2.34×10^{7}	0.05	1.17×10^{6}
Xenon-135	2.48×10^{7}	0.05	1.24×10^{6}
Cesium-134	3.79×10^{5}	5×10^{-4}	190

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Cesium-136	7.22×10^{5}	5×10^{-4}	361
Cesium-137	6.37×10^{5}	5×10^{-4}	318
Barium-139	1.99×10^{7}	5×10^{-4}	9,956
Barium-140	1.84×10^{7}	5×10^{-4}	9,191
Lanthanum-140	1.86×10^{7}	5×10^{-4}	9,281
Lanthanum-141	1.89×10^{7}	5×10^{-4}	9,442
Lanthanum-142	1.67×10^{7}	5×10^{-4}	8,367
Cerium-141	1.76×10^{7}	5×10^{-4}	8,781
Cerium-143	1.53×10^{7}	5×10^{-4}	7,670
Cerium-144	7.14×10^{6}	5×10^{-4}	3,570
Praseodymium-143	1.51×10^{7}	5×10^{-4}	7,536
Neodymium-147	7.55×10^{6}	5×10^{-4}	3,773
Rhenium-186	3.05×10^{4}	5×10^{-4}	18.2
Radium-223	235	5×10^{-4}	0.00
Radium-224	466	5×10^{-4}	0.00
Radium-226	172	5×10^{-4}	0.00
Actinium-227	408	5×10^{-4}	0.204
Thorium-227	298	5×10^{-4}	0.00
Thorium-228	505	5×10^{-4}	0.00
Neptunium-237	13.9	5×10^{-4}	0.00693
Neptunium-239	2.07×10^{8}	5×10^{-4}	1.04×10^{5}
Plutonium-238	7.12×10^{4}	5×10^{-4}	35.6
Plutonium-239	2.44×10^{4}	5×10^{-4}	12.2
Plutonium-240	1.97×10^{4}	5×10^{-4}	9.86
Plutonium-241	9.25×10^{5}	5×10^{-4}	463
Americium-241	1.07×10^{4}	5×10^{-4}	5.36
Curium-242	7.47×10^{5}	5×10^{-4}	384
Curium-244	631	5×10^{-4}	0.325

Source: Calculated results.

Table I–18 Beyond-Design-Basis Accident Source Term—Highly Enriched Uranium Fuel

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,980	0.05	249
Sodium-22	381	5.92×10^{-5}	0.0226
Sodium-24	5.00×10^{6}	5.92×10^{-5}	296
Cobalt-60	1.48×10^{7}	5×10^{-4}	7,400
Krypton-85	7.05×10^{4}	0.05	3,526
Krypton-85m	3.86×10^{6}	0.05	1.93×10^{5}
Krypton-87	7.28×10^{6}	0.05	3.64×10^{5}
Krypton-88	1.05×10^{7}	0.05	5.24×10^{5}
Rubidium-86	1.15×10^{5}	5×10^{-4}	57.7
Strontium-89	1.16×10^{7}	5×10^{-4}	5,797
Strontium-90	5.17×10^{5}	5×10^{-4}	259
Strontium-91	1.68×10^{7}	5×10^{-4}	8,381
Strontium-92	1.76×10^{7}	5×10^{-4}	8,825
Yttrium-90	5.74×10^{5}	5×10^{-4}	287
Yttrium-91	1.41×10^{7}	5×10^{-4}	7,051
Yttrium-92	1.77×10^{7}	5×10^{-4}	8,872
Yttrium-93	1.98×10^{7}	5×10^{-4}	9,896
Zirconium-95	1.71×10^{7}	5×10^{-4}	8,537
Zirconium-97	2.01×10^{7}	5×10^{-4}	1.00×10^{4}
Niobium-95	1.44×10^{7}	5×10^{-4}	7,202
Molybdenum-99	2.07×10^{7}	5×10^{-4}	1.03×10^{4}
Technetium-99m	1.81×10^{7}	5×10^{-4}	9,051
Ruthenium-103	1.39×10^{7}	5×10^{-4}	6,938
Ruthenium-105	8.40×10^{6}	5×10^{-4}	4,199
Ruthenium-106	2.52×10^{6}	5×10^{-4}	1,260
Rhodium-105	8.39×10^{6}	5×10^{-4}	4,194
Antimony-127	1.69×10^{6}	5×10^{-4}	846
Antimony-129	4.13×10^{6}	5×10^{-4}	2,064
Tellurium-127	1.62×10^{6}	5×10^{-4}	810
Tellurium-127m	1.65×10^{5}	5×10^{-4}	82.7
Tellurium-129	4.18×10^{6}	5×10^{-4}	2,088
Tellurium-129m	5.71×10^{5}	5×10^{-4}	285
Tellurium-131	1.09×10^{7}	5×10^{-4}	5,466
Tellurium-131m	1.60×10^{6}	5×10^{-4}	801
Tellurium-132	1.68×10^{7}	5×10^{-4}	8,389
Iodine-125	2,530	5×10^{-4}	1.30
Iodine-131	1.22×10^{7}	5×10^{-4}	6,096
Iodine-132	1.71×10^{7}	5×10^{-4}	8,531
Iodine-133	2.30×10^{7}	5×10^{-4}	1.15×10^{4}
Iodine-134	2.60×10^{7}	5×10^{-4}	1.30×10^{4}
Iodine-135	2.13×10^{7}	5×10^{-4}	1.06×10^{4}
Xenon-133	2.33×10^{7}	0.05	1.16×10^{6}
Xenon-135	2.29×10^{7}	0.05	1.15×10^{6}
Cesium-134	3.58×10^{5}	5×10^{-4}	179

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Cesium-136	4.32×10^{5}	5×10^{-4}	216
Cesium-137	6.34×10^{5}	5×10^{-4}	317
Barium-139	2.16×10^{7}	5×10^{-4}	1.08×10^{4}
Barium-140	2.03×10^{7}	5×10^{-4}	1.02×10^{4}
Lanthanum-140	2.05×10^{7}	5×10^{-4}	1.03×10^{4}
Lanthanum-141	2.03×10^{7}	5×10^{-4}	1.01×10^{4}
Lanthanum-142	1.87×10^{7}	5×10^{-4}	9,354
Cerium-141	1.89×10^{7}	5×10^{-4}	9,474
Cerium-143	1.86×10^{7}	5×10^{-4}	9,312
Cerium-144	9.53×10^{6}	5×10^{-4}	4,763
Praseodymium-143	1.84×10^{7}	5×10^{-4}	9,184
Neodymium-147	8.34×10^{6}	5×10^{-4}	4,170
Rhenium-186	3.64×10^{4}	5×10^{-4}	18.2
Radium-223	235	5×10^{-4}	0.00
Radium-224	466	5×10^{-4}	0.00
Radium-226	172	5×10^{-4}	0.00
Actinium-227	408	5×10^{-4}	0.204
Thorium-227	298	5×10^{-4}	0.00
Thorium-228	505	5×10^{-4}	0.00
Neptunium-237	15.0	5×10^{-4}	0.00752
Neptunium-239	1.85×10^{8}	5×10^{-4}	9.24×10^{4}
Plutonium-238	6.66×10^{4}	5×10^{-4}	33.3
Plutonium-239	6,760	5×10^{-4}	3.38
Plutonium-240	2,300	5×10^{-4}	1.15
Plutonium-241	8.52×10^{4}	5×10^{-4}	42.6
Americium-241	853	5×10^{-4}	0.426
Curium-242	6.08×10^{4}	5×10^{-4}	30.4
Curium-244	51.3	5×10^{-4}	0.026

Source: Calculated results.

FUEL- AND TARGET-HANDLING ACCIDENTS

A range of accidents related to ex-reactor irradiated fuel- and target-handling were postulated to occur outside of the reactor vessel (i.e., nonreactor accidents). The accident scenarios were selected from the *FFTF Final Safety Analysis Report* (Dautel 2000) and evaluated using the existing FFTF irradiated-fuel source term and new source terms for the neptunium-237 and medical isotope targets. The consequences of ex-reactor accidents involving industrial and civilian nuclear energy research and development targets are expected to be bounded by the accident selected in this NI PEIS.

The accident that would lead to the maximum radiological consequences is a seismic event during fuel assembly transfer. The bottom-loading transfer cask is used to transfer single core components from the containment building to the sodium storage vessel located in the Fuel Storage Facility or to a cask at the cask-loading station in the Reactor Service Building. The bottom-loading transfer cask is qualified to protect a fuel element from breach of cladding during a design-basis earthquake. However, if an element is being transferred into or out of another vessel when a design-basis earthquake occurs, a potential for damage to the component exists. This event is much less likely than the design-basis earthquake because of the small fraction of process time spent in the transfer of an assembly from one vessel to another.

Although the bottom-loading transfer cask is designed to remain upright during a design-basis earthquake at all transfer locations, it could move along the supporting rails during such an event. The probability of a design-basis earthquake is about 1×10^{-4} per year. If an assembly were being transferred through the interface between the bottom-loading transfer cask and the top of the other vessel or a floor valve at the exact moment of a design-basis earthquake, then the movement of the bottom-loading transfer cask could produce bending stresses on the assembly. The likelihood of such an occurrence is on the order of 0.001 per year for the proposed mission, resulting in a combined frequency of 1×10^{-7} for this scenario. Failure of fuel pin cladding as a result of assembly bending is not predicted by analysis. However, for the purpose of showing the depth of protection provided by FFTF against any undue risk to the public health and safety, the conservative assumptions listed below for an extreme beyond-design-basis fuel-handling accident were made in the final safety analysis report and are specified for this reevaluation of a fuel assembly.

- The fuel region of the assembly is in the transfer interface, such that the fuel could be damaged.
- Five percent of the fuel pins are assumed to lose cladding integrity.
- Release fractions are 1.0 for tritium and noble gases, 0.5 for halogens, and 0.05 for volatile solids.
- The release fraction for transuranics and nonvolatile solids is determined as follows: 5 percent of the fuel in the column is crushed and 5 percent of the crushed fuel is of respirable size (equal to or less than 10 microns). A suspension and release fraction of 1 percent is assumed for the respirable particles, i.e., 1 percent is released from the bottom-loading transfer cask and from containment or the Reactor Service Building.
- A 50 percent plateout fraction is assumed for halogens.
- No containment isolation is assumed, and the release is assumed to occur at ground level.

In addition to the mixed oxide and highly enriched uranium fuel assemblies, this accident was analyzed for maximum releases from the neptunium-237 and worst-case medical, industrial, and research and development isotope targets. Because the medical, industrial, and research and development isotope target assemblies have not been structurally analyzed for this type of impact event, all the target assembly rods are assumed to breach. No credible scenario has been identified that could produce temperatures high enough to vaporize target material. Because only one assembly can be accommodated by the bottom-loading transfer cask, the maximum release for this accident is from one fuel or target assembly only.

The radioisotope inventory, release fractions, and resulting environmental release for the mixed oxide and highly enriched uranium assemblies are provided in **Tables I–19 and I–20**.

Table I–19 Mixed Oxide Driver Fuel Assembly Source Term

Radioisotope	MOX Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	57.56	0.05	2.88
Krypton-85	517.6	0.05	25.9
Krypton-85m	2.850×10^{4}	0.05	1,430
Krypton-87	4.716×10^{4}	0.05	2,360
Krypton-88	6.567×10^{4}	0.05	3,280
Rubidium-86	907.6	1.25×10^{-6}	0.00113
Strontium-89	7.598×10^{4}	1.25×10^{-6}	0.0950
Strontium-90	3.181×10^{3}	1.25×10^{-6}	0.00398
Strontium-91	1.205×10^{5}	1.25×10^{-6}	0.151
Strontium-92	1.418×10^{5}	1.25×10^{-6}	0.177
Yttrium-90	3,561	1.25×10^{-6}	0.00445
Yttrium-91	9.984×10^{4}	1.25×10^{-6}	0.125
Yttrium-92	1.434×10^{5}	1.25×10^{-6}	0.179
Yttrium-93	1.785×10^{5}	1.25×10^{-6}	0.223
Zirconium-95	1.776×10^{5}	1.25×10^{-6}	0.222
Zirconium-97	2.357×10^{5}	1.25×10^{-6}	0.295
Niobium-95	1.492×10^{5}	1.25×10^{-6}	0.187
Molybdenum-99	2.690×10^{5}	1.25×10^{-6}	0.336
Technetium-99m	2.355×10^{5}	1.25×10^{-6}	0.294
Ruthenium-103	2.718×10^{5}	1.25×10^{-6}	0.340
Ruthenium-105	2.261×10^{5}	1.25×10^{-6}	0.283
Ruthenium-106	9.408×10^{4}	1.25×10^{-6}	0.118
Rhodium-105	2.246×10^{5}	1.25×10^{-6}	0.281
Antimony-127	2.579×10^{4}	1.25×10^{-6}	0.0322
Antimony-129	6.280×10^{4}	1.25×10^{-6}	0.0785
Tellurium-127	2.471×10^{4}	1.25×10^{-6}	0.0309
Tellurium-127m	2,535	1.25×10^{-6}	0.00317
Tellurium-129	6.211×10^{4}	1.25×10^{-6}	0.0776
Tellurium-129m	8,626	1.25×10^{-6}	0.0108
Tellurium-131	1.546×10^{5}	1.25×10^{-6}	0.193
Tellurium-131m	2.684×10^{4}	1.25×10^{-6}	0.0336
Tellurium-132	2.314×10^{5}	1.25×10^{-6}	0.289
Iodine-131	1.759×10^{5}	0.0125	2,200
Iodine-132	2.367×10^{5}	0.0125	2,960
Iodine-133	2.996×10^{5}	0.0125	3,750
Iodine-134	3.173×10^{5}	0.0125	3,970
Iodine-135	2.883×10^{5}	0.0125	3,600
Xenon-133	3.051×10^{5}	0.05	1.53×10^{4}
Xenon-135	3.254×10^{5}	0.05	1.63×10^{4}
Cesium-134	4,980	0.00250	12.5
Cesium-136	9,451	0.00250	23.6
Cesium-137	8,361	0.00250	20.9
Barium-139	2.594×10^{5}	1.25×10^{-6}	0.324
Barium-140	2.397×10^{5}	1.25×10^{-6}	0.300
Lanthanum-140	2.421×10^{5}	1.25×10^{-6}	0.303
Lanthanum-141	2.460×10^{5}	1.25×10^{-6}	0.308
Lanthanum-142	2.179×10^{5}	1.25×10^{-6}	0.272
Cerium-141	2.294×10^{5}	1.25×10^{-6}	0.287

Radioisotope	MOX Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Cerium-143	1.998×10^{5}	1.25×10^{-6}	0.250
Cerium-144	9.360×10^{4}	1.25×10^{-6}	0.117
Praseodymium-143	1.966×10^{5}	1.25×10^{-6}	0.246
Neodymium-147	9.847×10^{4}	1.25×10^{-6}	0.123
Neptunium-237	9.117×10^{-3}	1.25×10^{-6}	1.14×10^{-8}
Neptunium-239	2.723×10^{6}	1.25×10^{-6}	3.40
Plutonium-238	123.6	1.25×10^{-6}	1.55×10^{-4}
Plutonium-239	320.1	1.25×10^{-6}	4.00×10^{-4}
Plutonium-240	259.3	1.25×10^{-6}	3.24×10^{-4}
Plutonium-241	1.213×10^{4}	1.25×10^{-6}	0.0152
Americium-241	141.1	1.25×10^{-6}	1.76×10^{-4}
Curium-242	9,829	1.25×10^{-6}	0.0123
Curium-244	8.305	1.25×10^{-6}	1.04×10^{-5}

Key: MOX, mixed oxide.
Source: Nielsen 1999; Wootan 1999.

Table I–20 Highly Enriched Uranium Driver Fuel Assembly Source Term

Radioisotope	HEU Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	66.04	0.0500	3.302
Krypton-85	962.1	0.0500	48.11
Krypton-85m	5.236×10^{4}	0.0500	2,618
Krypton-87	9.952×10^{4}	0.0500	4,976
Krypton-88	1.433×10^{5}	0.0500	7,165
Rubidium-86	1,571	1.25×10^{-6}	0.001964
Strontium-89	1.586×10^{5}	1.25×10^{-6}	0.1983
Strontium-90	7,107	1.25×10^{-6}	0.008884
Strontium-91	2.279×10^{5}	1.25×10^{-6}	0.2849
Strontium-92	2.385×10^{5}	1.25×10^{-6}	0.2981
Yttrium-90	7,885	1.25×10^{-6}	0.009856
Yttrium-91	1.922×10^{5}	1.25×10^{-6}	0.2403
Yttrium-92	2.397×10^{5}	1.25×10^{-6}	0.2996
Yttrium-93	2.655×10^{5}	1.25×10^{-6}	0.3319
Zirconium-95	2.274×10^{5}	1.25×10^{-6}	0.2843
Zirconium-97	2.636×10^{5}	1.25×10^{-6}	0.3295
Niobium-95	1.921×10^{5}	1.25×10^{-6}	0.2401
Molybdenum-99	2.693×10^{5}	1.25×10^{-6}	0.3366
Technetium-99m	2.358×10^{5}	1.25×10^{-6}	0.2948
Ruthenium-103	1.727×10^{5}	1.25×10^{-6}	0.2159
Ruthenium-105	9.789×10^{4}	1.25×10^{-6}	0.1224
Ruthenium-106	2,729	1.25×10^{-6}	0.03411
Rhodium-105	9.844×10^{4}	1.25×10^{-6}	0.1231
Antimony-127	2.172×10^{4}	1.25×10^{-6}	0.02715
Antimony-129	5.288×10^{4}	1.25×10^{-6}	0.06610
Tellurium-127	2.082×10^{4}	1.25×10^{-6}	0.02603
Tellurium-127m	2,134	1.25×10^{-6}	0.002668
Tellurium-129	5.366×10^{4}	1.25×10^{-6}	0.06708
Tellurium-129m	7,338	1.25×10^{-6}	0.009173
Tellurium-131	1.413×10^{5}	1.25×10^{-6}	0.1766
Tellurium-131m	2.028×10^{4}	1.25×10^{-6}	0.02535
Tellurium-132	2.174×10^{5}	1.25×10^{-6}	0.2718
Iodine-131	1.574×10^{5}	0.0125	1,968
Iodine-132	2.209×10^{5}	0.0125	2,761
Iodine-133	2.996×10^{5}	0.0125	3,745
Iodine-134	3.412×10^{5}	0.0125	4,265
Iodine-135	2.763×10^{5}	0.0125	3,454
Xenon-133	3.034×10^{5}	0.0500	1.517×10^{4}
Xenon-135	2.995×10^{5}	0.0500	1.498×10^{4}
Cesium-134	4,676	0.00250	11.69
Cesium-136	5,314	0.00250	13.29
Cesium-137	8,319	0.00250	20.80
Barium-139	2.833×10^{5}	1.25×10^{-6}	0.3541
Barium-140	2.674×10^{5}	1.25×10^{-6}	0.3343
Lanthanum-140	2.698×10^{5}	1.25×10^{-6}	0.3373
Lanthanum-141	2.658×10^{5}	1.25×10^{-6}	0.3323
Lanthanum-142	2.461×10^{5}	1.25×10^{-6}	0.3076
Cerium-141	2.492×10^{5}	1.25×10^{-6}	0.3115

Radioisotope	HEU Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Cerium-143	2.467×10^{5}	1.25×10^{-6}	0.3084
Cerium-144	1.277×10^{5}	1.25×10^{-6}	0.1596
Praseodymium-143	2.437×10^{5}	1.25×10^{-6}	0.3046
Neodymium-147	1.098×10^{5}	1.25×10^{-6}	0.1373
Neodymium-237	0.02577	1.25×10^{-6}	3.221×10^{-8}
Neptunium-239	2.406×10^{6}	1.25×10^{-6}	3.008
Plutonium-238	57.38	1.25×10^{-6}	7.173×10^{-5}
Plutonium-239	68.66	1.25×10^{-6}	8.583×10^{-5}
Plutonium-240	10.45	1.25×10^{-6}	1.306×10^{-5}
Plutonium-241	132.2	1.25×10^{-6}	1.653×10^{-4}
Americium-241	0.08854	1.25×10^{-6}	1.107×10^{-7}
Curium-242	2.844	1.25×10^{-6}	3.555×10^{-6}
Curium-244	9.215×10^{-4}	1.25×10^{-6}	1.152×10^{-9}

Key: HEU, highly enriched uranium.
Source: Nielsen 1999; Wootan 1999.

Each neptunium-237 target will contain 333 grams (11.7 ounces) of plutonium-238 (5,000 grams [176 ounces] per year divided by 15 targets per year). The release fractions are assumed to be the same as were used for the driver fuel assemblies. The radioisotope inventory, release fractions, and resulting environmental release for the neptunium-237 target assembly are provided in **Table I–21**.

Table I–21 Neptunium-237 Target Assembly Source Term

Radioisotope	Neptunium-237 Target Normalized to 1 Gram of Plutonium-238 Activity (curies)	Maximum Target Activity of 333 Grams of Plutonium-238 (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	0.00241	0.803	0.0500	0.0401
Krypton-85	0.0202	6.73	0.0500	0.336
Krypton-85m	5.30	1,760	0.0500	88.2
Krypton-87	8.83	2,940	0.0500	147
Krypton-88	12.4	4,130	0.0500	206
Rubidium-86	0.00762	2.54	1.25×10^{-6}	3.17×10^{-6}
Strontium-89	9.63	3,210	1.25×10^{-6}	0.00401
Strontium-90	0.127	42.3	1.25×10^{-6}	5.29×10^{-5}
Strontium-91	23.4	7,790	1.25×10^{-6}	0.00974
Strontium-92	28.4	9,460	1.25×10^{-6}	0.0118
Yttrium-90	0.128	42.6	1.25×10^{-6}	5.33×10^{-5}
Yttrium-91	13.2	4,400	1.25×10^{-6}	0.00549
Yttrium-92	28.7	9,560	1.25×10^{-6}	0.0119
Yttrium-93	37.2	1.24×10^{4}	1.25×10^{-6}	0.0155
Zirconium-95	24.7	8,230	1.25×10^{-6}	0.0103
Zirconium-97	51.0	1.70×10^{4}	1.25×10^{-6}	0.0212
Niobium-95	16.8	5,590	1.25×10^{-6}	0.00699
Molybdenum-99	56.3	1.87×10^{4}	1.25×10^{-6}	0.0234
Technetium-99m	50.1	1.67×10^{4}	1.25×10^{-6}	0.0209
Ruthenium-103	42.5	1.42×10^{4}	1.25×10^{-6}	0.0177
Ruthenium-105	51.7	1.72×10^{4}	1.25×10^{-6}	0.0215
Ruthenium-106	6.41	2,130	1.25×10^{-6}	0.00267
Rhodium-105	41.1	1.37×10^{4}	1.25×10^{-6}	0.0171
Antimony-127	4.44	1,480	1.25×10^{-6}	0.00185
Antimony-129	13.5	4,500	1.25×10^{-6}	0.00562
Tellurium-127	4.18	1,390	1.25×10^{-6}	0.00174
Tellurium-127m	0.243	80.9	1.25×10^{-6}	1.01×10^{-4}
Tellurium-129	12.9	4,300	1.25×10^{-6}	0.00537
Tellurium-129m	1.39	463	1.25×10^{-6}	5.79×10^{-4}
Tellurium-131	30.9	1.03×10^{4}	1.25×10^{-6}	0.0129
Tellurium-131m	5.96	1,980	1.25×10^{-6}	0.00248
Tellurium-132	47.1	1.57×10^{4}	1.25×10^{-6}	0.0196
Iodine-131	32.5	1.08×10^{4}	0.0125	135
Iodine-132	48.7	1.62×10^{4}	0.0125	203
Iodine-133	65.0	2.16×10^{4}	0.0125	271
Iodine-134	69.0	2.30×10^{4}	0.0125	287
Iodine-135	60.8	2.02×10^{4}	0.0125	253
Xenon-133	61.3	2.04×10^{4}	0.0500	1,020
Xenon-135	7.69	2,560	0.0500	128
Cesium-134	0.159	52.9	0.00250	0.132
Cesium-136	0.920	306	0.00250	0.766
Cesium-137	0.375	125	0.00250	0.312
Barium-139	54.1	1.80×10^{4}	1.25×10^{-6}	0.0225
Barium-140	45.1	1.50×10^{4}	1.25×10^{-6}	0.0188
Lanthanum-140	44.5	1.48×10^{4}	1.25×10^{-6}	0.0185
Lanthanum-141	51.3	1.71×10^{4}	1.25×10^{-6}	0.0214

Radioisotope	Neptunium-237 Target Normalized to 1 Gram of Plutonium-238 Activity (curies)	Maximum Target Activity of 333 Grams of Plutonium-238 (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-142	47.6	1.59×10^{4}	1.25×10^{-6}	0.0198
Praseodymium-143	35.6	1.19×10^{4}	1.25×10^{-6}	0.0148
Cerium-141	35.0	1.17×10^{4}	1.25×10^{-6}	0.0146
Cerium-143	42.5	1.42×10^{4}	1.25×10^{-6}	0.0177
Cerium-144	7.13	2,370	1.25×10^{-6}	0.00297
Neodymium-147	17.1	5,690	1.25×10^{-6}	0.00712
Neptunium-237	0.00360	1.20	1.25×10^{-6}	1.50×10^{-6}
Neptunium-239	16.8	5,590	1.25×10^{-6}	0.00699
Plutonium-238	16.9	5,630	1.25×10^{-6}	0.00703
Plutonium-239	0.00921	3.07	1.25×10^{-6}	3.83×10^{-6}
Plutonium-240	0.00393	1.31	1.25×10^{-6}	1.64×10^{-6}
Plutonium-241	0.853	284	1.25×10^{-6}	3.55×10^{-4}
Americium-241	0.00	0.00	1.25×10^{-6}	0.00
Curium-244	0.00	0.00	1.25×10^{-6}	0.00

Source: Nielsen 1999; Schnitzler 1999.

The bottom-loading transfer cask would be used to transfer the Long-Term Irradiation Vehicle medical isotope targets. Except for the xenon-127 product target, which has a gaseous target material (xenon-126), the chemical and physical forms of the target material have not been decided upon. The release mechanism is assumed to be a breaking or tearing of the cladding tube due to the impact of a heavy object. The recommended bounding airborne release fraction for powder in a can which is broken or torn open due to the impact of a heavy object is 0.001; the respirable fraction, 0.1 (DOE 1994a). This gives a net release fraction of 1.0×10^{-4} for nongases. The release fraction of gases is assumed to be 1.0. The Long-Term Irradiation Vehicle targets were screened using these release fractions, and it was determined that the actinium-227 product target would result in the maximum consequences. The complete radioisotope inventory, release fraction, and resulting environmental release are presented in **Table I–22**. Although the entire radioisotope content of the actinium-227 product target is presented, 98.6 percent of the consequences are attributable to actinium-227 and thorium-228. Over 99.9 percent of the consequences are attributable to six radioisotopes (actinium-227, radium-223, radium-224, radium-226, thorium-227, and thorium-228).

Table I–22 Actinium-227 Product Target Assembly Source Term

Radioisotope	Target Activity (curies)	Release Fraction	Environmental Release (curies)
Actinium-227	34.0	1.00×10^{-4}	0.00340
Actinium-228	56.1	1.00×10^{-4}	0.00561
Actinium-229	6.04×10^{-9}	1.00×10^{-4}	6.04×10^{-13}
Radium-226	14.3	1.00×10^{-4}	0.00143
Radium-227	4.23×10^{-7}	1.00×10^{-4}	4.23×10^{-11}
Radium-228	0.00101	1.00×10^{-4}	1.01×10^{-7}
Radium-229	5.00×10^{-14}	1.00×10^{-4}	5.00×10^{-18}
Thorium-227	24.8	1.00×10^{-4}	0.00248
Thorium-228	42.1	1.00×10^{-4}	0.00421
Thorium-229	8.63×10^{-4}	1.00×10^{-4}	8.63×10^{-8}
Actinium-225	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Astatine-217	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Bismuth-210	0.109	1.00×10^{-4}	1.09×10^{-5}
Bismuth-211	19.6	1.00×10^{-4}	0.00196
Bismuth-212	24.6	1.00×10^{-4}	0.00246
Bismuth-213	3.71×10^{-4}	1.00×10^{-4}	3.71×10^{-8}
Bismuth-214	14.3	1.00×10^{-4}	0.00143
Francium-221	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Francium-223	1.40×10^{-5}	1.00×10^{-4}	1.40×10^{-9}
Lead-209	3.69×10^{-4}	1.00×10^{-4}	3.69×10^{-8}
Lead-210	0.118	1.00×10^{-4}	1.18×10^{-5}
Lead-211	19.6	1.00×10^{-4}	0.00196
Lead-212	38.4	1.00×10^{-4}	0.00384
Lead-214	14.3	1.00×10^{-4}	0.00143
Polonium-210	0.106	1.00×10^{-4}	1.06×10^{-5}
Polonium-211	0.0535	1.00×10^{-4}	5.35×10^{-6}
Polonium-212	24.6	1.00×10^{-4}	0.00246
Polonium-213	3.63×10^{-4}	1.00×10^{-4}	3.63×10^{-8}
Polonium-214	14.3	1.00×10^{-4}	0.00143
Polonium-215	19.6	1.00×10^{-4}	0.00196
Polonium-216	38.8	1.00×10^{-4}	0.00388
Polonium-218	14.3	1.00×10^{-4}	0.00143
Radium-223	19.6	1.00×10^{-4}	0.00196
Radium-224	38.8	1.00×10^{-4}	0.00388
Radium-225	5.46×10^{-4}	1.00×10^{-4}	5.46×10^{-8}
Radon-217	4.46×10^{-8}	1.00	4.46×10^{-8}
Radon-219	19.6	1.00	19.6
Radon-220	38.8	1.00	38.8
Radon-222	14.3	1.00	14.3
Thallium-207	19.6	1.00×10^{-4}	0.00196
Thallium-208	8.83	1.00×10^{-4}	8.83×10^{-4}
Thallium-209	8.16×10^{-6}	1.00×10^{-4}	8.16×10^{-10}

Source: Nielsen 1999; BWHC 1999.

I.1.1.4.2 FFTF Standby

The limiting accident for FFTF in its current standby condition is a primary heat transport system sodium spill. This accident has a frequency of about 1×10^{-4} per year. In its standby condition, the FFTF primary sodium is far less radioactive than under the proposed operating conditions. This is mainly because the fuel has been previously removed, but also because the radioactive sodium has had time to decay.

The current radioactive inventory in the primary heat transport sodium is provided **Table I–23**.

Table I–23 Current FFTF Primary Sodium Activity

Isotope	Activity (curies)
Hydrogen-3	54.9
Sodium-22	76.0
Cesium-137	0.0384
Plutonium-239	5.07×10^{-4}

Source: Nielsen 2000.

The size of the sodium spill is equivalent to that of the design-basis accident (393 kilograms [867 pounds], of which 24.4 kilograms [53.7 pounds] burn). Since the reactor is in a standby condition, no credit is taken for containment holdup of releases. Therefore, the release fraction is simply the ratio of the sodium burned to the total sodium inventory (i.e., $53.7/930{,}220 = 5.77\times10^{-5}$). The FFTF standby accident source term is provided in **Table I–24**.

Table I–24 FFTF Standby Accident Source Term

Isotope	Environmental Release (curies)
Hydrogen-3	0.00317
Sodium-22	0.00439
Cesium-137	2.22×10^{-6}
Plutonium-239	2.93×10^{-8}

Source: Calculated results.

It should be noted that the radioactive isotopes are continuously reduced by radioactive decay. Examination of the current inventories and dose conversion factors for these isotopes reveals that almost the entire dose would be attributable to plutonium-239 and sodium-22. Plutonium-239 has an extremely long half-life (24,000 years) and therefore its rather small decay would have little effect on consequences for quite some time. Sodium-22, however, has a fairly short half-life (2.6 years), and its decay would have a significant effect on the dose. For instance, after 35 years, only 21 percent of the original dose level would remain.

I.1.1.4.3 FFTF Deactivation

The limiting deactivation accident was determined from a review of the *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington* (DOE 1995). The bounding accident is a sodium spill during the transfer of the primary sodium to a treatment tank. A 9.1-kilogram (20-pound) spill of primary sodium outside the containment is assumed. The release fractions are 100 percent for noble gases and 1 percent for nongases.

The resulting source term, based on current primary sodium radioactivity is presented in **Table I–25**. As noted in the standby accident, the primary sodium radioactivity is continuously being reduced by radioactive decay.

Table I–25 FFTF Deactivation Accident Source Term

Isotope	Environmental Release (curies)
Hydrogen-3	0.00118
Sodium-22	1.63×10^{-5}
Cesium-137	8.26×10^{-9}
Plutonium-239	1.09×10^{-10}

Source: Calculated results.

The environmental analysis states that the accident frequency is greater than 0.01. For this NI PEIS, the accident frequency is conservatively chosen to be 0.10. This frequency is the probability of a sodium spill during the sodium transfer process. It is a frequency per event rather than per year.

I.1.1.4.4 Meteorological Data

Meteorological characteristics of the FFTF site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at the Hanford 400 Area.

I.1.1.4.5 Population Data

The population distribution surrounding FFTF is based on the 1990 census (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.4.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for FFTF accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.5 Low-Energy Accelerator

A spectrum of potential accidents at a low-energy accelerator used for the production of medical, industrial, and research and development isotopes was investigated. The accidents with the greatest potential for onsite and offsite consequences were evaluated in detail.

I.1.1.5.1 Design-Basis Accident

The limiting design-basis accident at the low-energy accelerator was determined to be a target assembly handling accident with an estimated probability of 1.0×10^{-4} per year (TechSource 2000).

The accident is assumed to occur one day after the beam is shut off. The medical, industrial, or research and development target is assumed to be damaged from mishandling. One hundred percent of the volatile fission products is assumed to be released from the target into the building. One percent of all the nonvolatile radioisotopes are released into the building. Fifty percent of the released radioisotopes, except noble gases, are assumed to plateout in the building. The radioisotopes which do not plateout are released to the environment through two stages of high-efficiency particulate air (HEPA) filters with a 99.95 percent efficiency for each stage and an activated charcoal filter with an assumed 99 percent iodine removal efficiency.

These assumptions result in a release fraction of 1.25×10^{-9} ($0.01 \times 0.5 \times 0.0005 \times 0.0005$) for the nonvolatile radioisotopes, 0.005 ($1 \times 0.5 \times 0.01$) for iodine, and 1.0 for noble gases. The likely medical, industrial, and research and development targets were screened with these release fractions to determine which target would result in the highest consequences from the target-handling accident. The target with the highest consequence is the iodine-125 product target with an environmental release of 12.7 curies. The likely medical, industrial, and research and development target product inventories are provided in Section I.1.4.2.

I.1.1.5.2 Beyond-Design-Basis Accident

The beyond-design-basis accident for the low-energy accelerator is a severe earthquake with an estimated frequency of 1.0×10^{-5} per year (TechSource 2000).

The medical, industrial, or research and development target is assumed to be crushed. One hundred percent of the volatile fission products are assumed to be released from the target into the building. One percent of the nonvolatile radioisotopes are assumed to be released into the building. None of the noble gases, 50 percent of the iodine, and 90 percent of the other radioisotopes are assumed to plateout in the building. The HEPA and charcoal filters are assumed to be destroyed and ineffective.

These assumptions result in a release fraction of 0.001 (0.01×0.1) for the nonvolatile radioisotopes, 0.5 (1×0.5) for iodine, and 1.0 for noble gases. The likely medical, industrial, and research and development targets were screened with these release fractions to determine which target would result in the highest consequences from the severe earthquake accident. The target with the highest consequence is the actinium-227 product target with the source term presented in **Table I–26**.

Table I–26 Low-Energy Accelerator Beyond-Design-Basis Accident Source Term

Isotope	Target Product Inventory[a] (curies)	Release Fraction	Environmental Release (curies)
Actinium-227	3.40×10^{1}	1.0×10^{-3}	3.40×10^{-2}
Radium-223	1.96×10^{1}	1.0×10^{-3}	1.96×10^{-2}
Radium-224	3.88×10^{1}	1.0×10^{-3}	3.88×10^{-2}
Radium-226	1.43×10^{1}	1.0×10^{-3}	1.43×10^{-2}
Thorium-227	2.48×10^{1}	1.0×10^{-3}	2.48×10^{-2}
Thorium-228	4.21×10^{1}	1.0×10^{-3}	4.21×10^{-2}

a. Although the product target contains several other radioisotopes, these six radioisotopes contribute over 99.9 percent of the dose consequences.

I.1.1.5.3 Meteorological Data

The meteorological characteristics of the generic accelerator site are assumed to be the same as those for the generic CLWR site and are described in Section I.1.1.3.2.

I.1.1.5.4 Population Data

The population distribution surrounding the generic accelerator site is assumed to be the same as that for the generic CLWR site and is described in Section I.1.1.3.3.

I.1.1.5.5 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event

of an emergency. However, no relocation or evacuation of the offsite population was assumed for FFTF accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.6 High-Energy Accelerator

A spectrum of potential accidents at a high-energy accelerator used for the production of plutonium-238 was investigated. The accidents with the greatest potential for onsite and offsite consequences were evaluated in detail. The meteorological data, population data, and evacuation information for the high-energy accelerator analysis are the same as those used for the low-energy accelerator analysis.

I.1.1.6.1 Design-Basis Accident

The limiting design-basis accident for the high-energy proton accelerator was determined to be a target assembly handling accident with an estimated probability of 1.0×10^{-4} per year (TechSource 2000).

The target is exposed to the beam for an estimated 99 days. The accident is assumed to occur after the target has been exposed to the beam for the full 99 days and 1 day after the beam is shut off. The target assembly is postulated to be dropped and partially crushed as it is being moved from the beam location to a cooled storage well. Without cooling, the assembly is estimated to begin to melt in 70 minutes. It would take about 1 to 4 hours to retrieve a target assembly and place it in a cooled storage well. It is assumed that 2 hours pass before retrieving the assembly and that the target has begun to melt.

Secondary neutrons produce a number of fissions within the uranium-238 target and the neptunium-237 blanket. The target and blanket generate 2.464 and 0.164 megawatts, respectively, from fission. The quantity of fission products produced in the target and blanket assembly is equivalent to that of a low-power reactor operating at a power level of 2.63 megawatts for 99 days.

In addition to fission products, high-energy protons striking the target also produce spallation products that results in isotopes from several to many mass units lower than the original target nucleus. The heat generated by spallation products is estimated to be 3.32 megawatts (2.604 and 0.719 megawatts in the target and blanket, respectively). The resulting spallation product isotopes were not directly calculated. An estimate was made by noting that approximately one isotope chain results per spall, whereas two result per fission. Therefore, it was assumed that there are one-half as many isotopic chains for the spallation process and that these chains are also comparable to those found in nuclear reactors. Hence, the radionuclides generated by spallation were characterized as having the same composition and quantities as fission products from a reactor operating at a power level of one-half of 3.32, or 1.66 megawatts, for a period of 99 days. This is considered to be conservative since the fraction of spallation products having masses comparable to the more volatile and hazardous fission products such as iodine are estimated to be smaller than the fission product yield of these same isotopes.

The total activity in the target and blanket assembly consists of the target and blanket materials, plutonium-238 and other isotopes produced (e.g., beryllium-7), fission products, and spallation products.

As a result of the accident, 1 percent of all the radioisotopes is assumed to be released into the building. Fifty percent of the released radioisotopes, except noble gases, is assumed to plate out or deposit within the building. The radioisotopes that do not plate out are released to the environment after passing through two stages of HEPA filters with a 99.95 percent efficiency for each stage, and a charcoal filter with an iodine removal efficiency of 99 percent. This results in release fractions of 0.01 for the noble gases, 5×10^{-5} ($0.01 \times 0.5 \times 0.01$) for the iodines, and 1.25×10^{-9} ($0.01 \times 0.5 \times 0.0005 \times 0.0005$) for the nonvolatile radioisotopes. The release

is elevated from a 30-meter-high (98-foot-high) stack. The source term for the design-basis accident is presented in **Table I–27**.

Table I–27 Accelerator Design-Basis Accident Source Term

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Beryllium-7	3.60×10^{2}	-	-	3.60×10^{2}	1.25×10^{-9}	4.50×10^{-7}
Cobalt-58	–	4.15×10^{2}	2.62×10^{2}	6.77×10^{2}	1.25×10^{-9}	8.47×10^{-7}
Cobalt-60	–	7.78×10^{1}	4.91×10^{1}	1.27×10^{2}	1.25×10^{-9}	1.59×10^{-7}
Krypton-85	–	7.39×10^{1}	4.66×10^{1}	1.21×10^{2}	1.00×10^{-2}	1.21
Krypton-85m	–	2.41×10^{4}	1.52×10^{4}	3.94×10^{4}	1.00×10^{-2}	3.94×10^{2}
Krypton-87	–	4.41×10^{4}	2.78×10^{4}	7.19×10^{4}	1.00×10^{-2}	7.19×10^{2}
Krypton-88	–	5.97×10^{4}	3.77×10^{4}	9.73×10^{4}	1.00×10^{-2}	9.73×10^{2}
Rubidium-86	–	3.83×10^{1}	2.42×10^{1}	6.25×10^{1}	1.25×10^{-9}	7.81×10^{-8}
Strontium-89	–	5.48×10^{4}	3.46×10^{4}	8.94×10^{4}	1.25×10^{-9}	1.12×10^{-4}
Strontium-90	–	5.60×10^{2}	3.53×10^{2}	9.13×10^{2}	1.25×10^{-9}	1.14×10^{-6}
Strontium-91	–	9.64×10^{4}	6.08×10^{4}	1.57×10^{5}	1.25×10^{-9}	1.96×10^{-4}
Strontium-92	–	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	1.25×10^{-9}	2.04×10^{-4}
Yttrium-90	–	4.33×10^{3}	2.73×10^{3}	7.07×10^{3}	1.25×10^{-9}	8.83×10^{-6}
Yttrium-91	–	6.26×10^{4}	3.95×10^{4}	1.02×10^{5}	1.25×10^{-9}	1.28×10^{-4}
Yttrium-92	–	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	1.25×10^{-9}	2.04×10^{-4}
Yttrium-93	–	1.13×10^{5}	7.15×10^{4}	1.85×10^{5}	1.25×10^{-9}	2.31×10^{-4}
Zirconium-95	–	7.46×10^{4}	4.71×10^{4}	1.22×10^{5}	1.25×10^{-9}	1.52×10^{-4}
Zirconium-97	–	1.20×10^{5}	7.59×10^{4}	1.96×10^{5}	1.25×10^{-9}	2.45×10^{-4}
Niobium-95	–	9.33×10^{4}	5.89×10^{4}	1.52×10^{5}	1.25×10^{-9}	1.90×10^{-4}
Molybdenum-99	–	1.27×10^{5}	8.03×10^{4}	2.07×10^{5}	1.25×10^{-9}	2.59×10^{-4}
Technetium-99m	–	1.09×10^{5}	6.91×10^{4}	1.79×10^{5}	1.25×10^{-9}	2.23×10^{-4}
Ruthenium-103	–	7.80×10^{4}	4.93×10^{4}	1.27×10^{5}	1.25×10^{-9}	1.59×10^{-4}
Ruthenium-105	–	6.15×10^{4}	3.88×10^{4}	1.00×10^{5}	1.25×10^{-9}	1.25×10^{-4}
Ruthenium-106	–	4.89×10^{3}	3.09×10^{3}	7.98×10^{3}	1.25×10^{-9}	9.97×10^{-6}
Rhodium-105	–	4.26×10^{4}	2.69×10^{4}	6.95×10^{4}	1.25×10^{-9}	8.69×10^{-5}
Antimony-127	–	5.80×10^{3}	3.66×10^{3}	9.47×10^{3}	1.25×10^{-9}	1.18×10^{-5}
Antimony-129	–	2.06×10^{4}	1.30×10^{4}	3.36×10^{4}	1.25×10^{-9}	4.20×10^{-5}
Tellurium-127	–	5.61×10^{3}	3.54×10^{3}	9.15×10^{3}	1.25×10^{-9}	1.14×10^{-5}
Tellurium-127m	–	3.50×10^{2}	2.21×10^{2}	5.71×10^{2}	1.25×10^{-9}	7.14×10^{-7}
Tellurium-129	–	1.93×10^{4}	1.22×10^{4}	3.14×10^{4}	1.25×10^{-9}	3.93×10^{-5}
Tellurium-129m	–	4.43×10^{3}	2.80×10^{3}	7.23×10^{3}	1.25×10^{-9}	9.04×10^{-6}
Tellurium-131m	–	9.71×10^{3}	6.13×10^{3}	1.58×10^{4}	1.25×10^{-9}	1.98×10^{-5}
Tellurium-132	–	9.71×10^{4}	6.13×10^{4}	1.58×10^{5}	1.25×10^{-9}	1.98×10^{-4}
Iodine-131	–	6.67×10^{4}	4.21×10^{4}	1.09×10^{5}	5.00×10^{-5}	5.44×10^{1}
Iodine-132	–	9.87×10^{4}	6.23×10^{4}	1.61×10^{5}	5.00×10^{-5}	8.05×10^{1}
Iodine-133	–	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-5}	1.15×10^{2}
Iodine-134	–	1.55×10^{5}	9.78×10^{4}	2.53×10^{5}	5.00×10^{-5}	1.26×10^{2}
Iodine-135	–	1.33×10^{5}	8.42×10^{4}	2.18×10^{5}	5.00×10^{-5}	1.09×10^{2}
Xenon-133	–	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	1.00×10^{-2}	2.30×10^{3}
Xenon-135	–	2.65×10^{4}	1.67×10^{4}	4.33×10^{4}	1.00×10^{-2}	4.33×10^{2}
Cesium-134	–	1.61×10^{3}	1.01×10^{3}	2.62×10^{3}	1.25×10^{-9}	3.27×10^{-6}
Cesium-136	–	2.73×10^{3}	1.72×10^{3}	4.45×10^{3}	1.25×10^{-9}	5.57×10^{-6}
Cesium-137	–	6.96×10^{2}	4.39×10^{2}	1.14×10^{3}	1.25×10^{-9}	1.42×10^{-6}
Barium-139	–	1.31×10^{5}	8.27×10^{4}	2.14×10^{5}	1.25×10^{-9}	2.67×10^{-4}

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Barium-140	–	1.29×10^{5}	8.14×10^{4}	2.10×10^{5}	1.25×10^{-9}	2.63×10^{-4}
Lanthanum-140	–	1.33×10^{5}	8.37×10^{4}	2.16×10^{5}	1.25×10^{-9}	2.70×10^{-4}
Lanthanum-141	–	1.21×10^{5}	7.64×10^{4}	1.97×10^{5}	1.25×10^{-9}	2.47×10^{-4}
Lanthanum-142	–	1.17×10^{5}	7.40×10^{4}	1.91×10^{5}	1.25×10^{-9}	2.39×10^{-4}
Cerium-141	–	1.04×10^{5}	6.54×10^{4}	1.69×10^{5}	1.25×10^{-9}	2.11×10^{-4}
Cerium-143	–	1.14×10^{5}	7.20×10^{4}	1.86×10^{5}	1.25×10^{-9}	2.33×10^{-4}
Cerium-144	–	1.83×10^{4}	1.15×10^{4}	2.98×10^{4}	1.25×10^{-9}	3.73×10^{-5}
Praseodymium-143	–	1.12×10^{5}	7.06×10^{4}	1.82×10^{5}	1.25×10^{-9}	2.28×10^{-4}
Neodymium-147	–	5.02×10^{4}	3.17×10^{4}	8.18×10^{4}	1.25×10^{-9}	1.02×10^{-4}
Neptunium-237	4.93×10^{1}	–	–	4.93×10^{1}	1.25×10^{-9}	6.16×10^{-8}
Neptunium-238	1.41×10^{7}	–	–	1.41×10^{7}	1.25×10^{-9}	1.76×10^{-2}
Neptunium-239	3.01×10^{6}	1.35×10^{6}	8.51×10^{5}	5.21×10^{6}	1.25×10^{-9}	6.51×10^{-3}
Plutonium-238	3.40×10^{4}	1.04×10^{1}	6.58	3.40×10^{4}	1.25×10^{-9}	4.25×10^{-5}
Plutonium-239	2.69×10^{1}	2.33	1.47	3.07×10^{1}	1.25×10^{-9}	3.84×10^{-8}
Plutonium-240	–	2.95	1.86	4.81	1.25×10^{-9}	6.01×10^{-9}
Plutonium-241	–	5.16×10^{2}	3.26×10^{2}	8.42×10^{2}	1.25×10^{-9}	1.05×10^{-6}
Americium-241	–	3.28×10^{-1}	2.07×10^{-1}	5.34×10^{-1}	1.25×10^{-9}	6.68×10^{-10}
Curium-242	–	3.33×10^{2}	2.10×10^{2}	5.43×10^{2}	1.25×10^{-9}	6.78×10^{-7}
Curium-244	–	7.58	4.79	1.24×10^{1}	1.25×10^{-9}	1.55×10^{-8}

I.1.1.6.2 Beyond-Design-Basis Accident

The beyond-design-basis accident for the high-energy accelerator is a severe earthquake with an estimated frequency of 1.0×10^{-5} per year (TechSource 2000). The earthquake is postulated to occur on the 99th day of beam irradiation, rupturing all cooling pipes serving the target assembly. Without cooling, the target assembly is estimated to begin to melt within about 30 minutes.

Five percent of the radioisotopes are assumed to be released into the building. The ventilation systems fail, resulting in a slow radioisotope transport from the building. Because of the slow transport, 90 percent of the radioisotopes (except for the noble gases) are assumed to plate out in the building. It is assumed that the earthquake has also destroyed the HEPA and charcoal filters as well as the stack. These assumptions result in a ground level release with release fractions of 0.05 (0.05 × 1) for the noble gases, and 0.005 (0.05 × 0.1) for the others. The source term for the beyond-design-basis accident is presented in **Table I–28**.

Table I–28 Accelerator Beyond-Design-Basis Accident Source Term

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Beryllium-7	3.60×10^{2}	-	-	3.60×10^{2}	5.00×10^{-3}	1.80
Cobalt-58	-	4.15×10^{2}	2.62×10^{2}	6.77×10^{2}	5.00×10^{-3}	3.39
Cobalt-60	-	7.78×10^{1}	4.91×10^{1}	1.27×10^{2}	5.00×10^{-3}	6.34×10^{-1}
Krypton-85	-	7.39×10^{1}	4.66×10^{1}	1.21×10^{2}	5.00×10^{-2}	6.03
Krypton-85m	-	2.41×10^{4}	1.52×10^{4}	3.94×10^{4}	5.00×10^{-2}	1.97×10^{3}
Krypton-87	-	4.41×10^{4}	2.78×10^{4}	7.19×10^{4}	5.00×10^{-2}	3.60×10^{3}
Krypton-88	-	5.97×10^{4}	3.77×10^{4}	9.73×10^{4}	5.00×10^{-2}	4.87×10^{3}
Rubidium-86	-	3.83×10^{1}	2.42×10^{1}	6.25×10^{1}	5.00×10^{-3}	3.13×10^{-1}
Strontium-89	-	5.48×10^{4}	3.46×10^{4}	8.94×10^{4}	5.00×10^{-3}	4.47×10^{2}
Strontium-90	-	5.60×10^{2}	3.53×10^{2}	9.13×10^{2}	5.00×10^{-3}	4.56
Strontium-91	-	9.64×10^{4}	6.08×10^{4}	1.57×10^{5}	5.00×10^{-3}	7.86×10^{2}
Strontium-92	-	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	5.00×10^{-3}	8.17×10^{2}
Yttrium-90	-	4.33×10^{3}	2.73×10^{3}	7.07×10^{3}	5.00×10^{-3}	3.53×10^{1}
Yttrium-91	-	6.26×10^{4}	3.95×10^{4}	1.02×10^{5}	5.00×10^{-3}	5.11×10^{2}
Yttrium-92	-	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	5.00×10^{-3}	8.17×10^{2}
Yttrium-93	-	1.13×10^{5}	7.15×10^{4}	1.85×10^{5}	5.00×10^{-3}	9.24×10^{2}
Zirconium-95	-	7.46×10^{4}	4.71×10^{4}	1.22×10^{5}	5.00×10^{-3}	6.08×10^{2}
Zirconium-97	-	1.20×10^{5}	7.59×10^{4}	1.96×10^{5}	5.00×10^{-3}	9.81×10^{2}
Niobium-95	-	9.33×10^{4}	5.89×10^{4}	1.52×10^{5}	5.00×10^{-3}	7.61×10^{2}
Molybdenum-99	-	1.27×10^{5}	8.03×10^{4}	2.07×10^{5}	5.00×10^{-3}	1.04×10^{3}
Technetium-99m	-	1.09×10^{5}	6.91×10^{4}	1.79×10^{5}	5.00×10^{-3}	8.93×10^{2}
Ruthenium-103	-	7.80×10^{4}	4.93×10^{4}	1.27×10^{5}	5.00×10^{-3}	6.37×10^{2}
Ruthenium-105	-	6.15×10^{4}	3.88×10^{4}	1.00×10^{5}	5.00×10^{-3}	5.02×10^{2}
Ruthenium-106	-	4.89×10^{3}	3.09×10^{3}	7.98×10^{3}	5.00×10^{-3}	3.99×10^{1}
Rhodium-105	-	4.26×10^{4}	2.69×10^{4}	6.95×10^{4}	5.00×10^{-3}	3.48×10^{2}
Antimony-127	-	5.80×10^{3}	3.66×10^{3}	9.47×10^{3}	5.00×10^{-3}	4.73×10^{1}
Antimony-129	-	2.06×10^{4}	1.30×10^{4}	3.36×10^{4}	5.00×10^{-3}	1.68×10^{2}
Tellurium-127	-	5.61×10^{3}	3.54×10^{3}	9.15×10^{3}	5.00×10^{-3}	4.58×10^{1}
Tellurium-127m	-	3.50×10^{2}	2.21×10^{2}	5.71×10^{2}	5.00×10^{-3}	2.86
Tellurium-129	-	1.93×10^{4}	1.22×10^{4}	3.14×10^{4}	5.00×10^{-3}	1.57×10^{2}
Tellurium-129m	-	4.43×10^{3}	2.80×10^{3}	7.23×10^{3}	5.00×10^{-3}	3.62×10^{1}
Tellurium-131m	-	9.71×10^{3}	6.13×10^{3}	1.58×10^{4}	5.00×10^{-3}	7.92×10^{1}
Tellurium-132	-	9.71×10^{4}	6.13×10^{4}	1.58×10^{5}	5.00×10^{-3}	7.92×10^{2}
Iodine-131	-	6.67×10^{4}	4.21×10^{4}	1.09×10^{5}	5.00×10^{-3}	5.44×10^{2}
Iodine-132	-	9.87×10^{4}	6.23×10^{4}	1.61×10^{5}	5.00×10^{-3}	8.05×10^{2}
Iodine-133	-	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-3}	1.15×10^{3}
Iodine-134	-	1.55×10^{5}	9.78×10^{4}	2.53×10^{5}	5.00×10^{-3}	1.26×10^{3}
Iodine-135	-	1.33×10^{5}	8.42×10^{4}	2.18×10^{5}	5.00×10^{-3}	1.09×10^{3}
Xenon-133	-	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-2}	1.15×10^{4}
Xenon-135	-	2.65×10^{4}	1.67×10^{4}	4.33×10^{4}	5.00×10^{-2}	2.16×10^{3}
Cesium-134	-	1.61×10^{3}	1.01×10^{3}	2.62×10^{3}	5.00×10^{-3}	1.31×10^{1}
Cesium-136	-	2.73×10^{3}	1.72×10^{3}	4.45×10^{3}	5.00×10^{-3}	2.23×10^{1}
Cesium-137	-	6.96×10^{2}	4.39×10^{2}	1.14×10^{3}	5.00×10^{-3}	5.68
Barium-139	-	1.31×10^{5}	8.27×10^{4}	2.14×10^{5}	5.00×10^{-3}	1.07×10^{3}
Barium-140	-	1.29×10^{5}	8.14×10^{4}	2.10×10^{5}	5.00×10^{-3}	1.05×10^{3}

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-140	-	1.33×10^{5}	8.37×10^{4}	2.16×10^{5}	5.00×10^{-3}	1.08×10^{3}
Lanthanum-141	-	1.21×10^{5}	7.64×10^{4}	1.97×10^{5}	5.00×10^{-3}	9.87×10^{2}
Lanthanum-142	-	1.17×10^{5}	7.40×10^{4}	1.91×10^{5}	5.00×10^{-3}	9.56×10^{2}
Cerium-141	-	1.04×10^{5}	6.54×10^{4}	1.69×10^{5}	5.00×10^{-3}	8.45×10^{2}
Cerium-143	-	1.14×10^{5}	7.20×10^{4}	1.86×10^{5}	5.00×10^{-3}	9.30×10^{2}
Cerium-144	-	1.83×10^{4}	1.15×10^{4}	2.98×10^{4}	5.00×10^{-3}	1.49×10^{2}
Praseodymium-143	-	1.12×10^{5}	7.06×10^{4}	1.82×10^{5}	5.00×10^{-3}	9.12×10^{2}
Neodymium-147	-	5.02×10^{4}	3.17×10^{4}	8.18×10^{4}	5.00×10^{-3}	4.09×10^{2}
Neptunium-237	4.93×10^{1}	-	-	4.93×10^{1}	5.00×10^{-3}	2.47×10^{-1}
Neptunium-238	1.41×10^{7}	-	-	1.41×10^{7}	5.00×10^{-3}	7.05×10^{4}
Neptunium-239	3.01×10^{6}	1.35×10^{6}	8.51×10^{5}	5.21×10^{6}	5.00×10^{-3}	2.61×10^{4}
Plutonium-238	3.40×10^{4}	1.04×10^{1}	6.58	3.40×10^{4}	5.00×10^{-3}	1.70×10^{2}
Plutonium-239	2.69×10^{1}	2.33	1.47	3.07×10^{1}	5.00×10^{-3}	1.54×10^{-1}
Plutonium-240	-	2.95	1.86	4.81	5.00×10^{-3}	2.40×10^{-2}
Plutonium-241	-	5.16×10^{2}	3.26×10^{2}	8.42×10^{2}	5.00×10^{-3}	4.21
Americium-241	-	3.28×10^{-1}	2.07×10^{-1}	5.34×10^{-1}	5.00×10^{-3}	2.67×10^{3}
Curium-242	-	3.33×10^{2}	2.10×10^{2}	5.43×10^{2}	5.00×10^{-3}	2.71
Curium-244	-	7.58	4.79	1.24×10^{1}	5.00×10^{-3}	6.18×10^{-2}

I.1.1.7 New Research Reactor

The new research reactor would produce a number of long- and short-lived isotopes for medical and industrial applications and 5 kilograms (11 pounds) of plutonium-238 per year for space power applications. The new research reactor would contain 48 neptunium-237 target assemblies, each assembly consisting of four neptunium-237 target rods. The maximum plutonium-238 produced in each target rod is 27.6 grams (0.97 ounce). The reactor would also contain eight medical and industrial target assemblies, each assembly consisting of two medical and industrial target rods. The reactor would also contain eight rabbit tubes for short-irradiation-time production of medical or industrial isotopes and civilian nuclear energy research and development. The rabbit tubes are outside the fuel region of the core, but still within an area with a rather high flux. Detailed descriptions of the new research reactor are provided in Appendix E. The meteorological data, population data, and evacuation information for the new research reactor analysis are the same as those used for the low-energy accelerator analysis.

I.1.1.7.1 Design-Basis Accident

A spectrum of accidents was reviewed according to the guidance provided in NUREG-1537 (NRC 1996). It was concluded that the maximum hypothetical accident is an accident whose potential consequences would exceed and bound all credible accidents. The accident scenario was assumed to represent the design-basis accident for the new research reactor.

Operational incidents leading to loss of coolant, loss of flow, loss of normal electrical power, and reactivity insertion would not result in any fuel damage. The built-in safety features of the new research reactor, such as elevation of the spent fuel pool system above the core, elevation of primary piping above the core, and antisyphon devices, would preclude loss of core cooling capability. The inherent large prompt negative fuel temperature coefficient of reactivity, would minimize the effect of accidental reactivity insertion. The reactivity insertion would cause a sudden increase in reactor power, leading to a higher fuel temperature which,

in turn, because of its large negative temperature coefficient of reactivity, would shut down the reactor. The design of the control rods would limit reactivity insertion below that which could cause any fuel failure.

The design-basis accident for the reactor is cladding failure of a single irradiated element (NRC 1996). The single fuel element could fail due to material deficiencies at any time during normal operation or while the reactor is shut down. Judging from experience with TRIGA (training, research, isotopes General Atomics) fuels, this type of failure is considered infrequent (a likelihood of 1 in 100 years) (UC-Davis 1999). For the new research reactor, it was assumed that the cladding of all fuel rods in an assembly (a maximum of 64 rods) would fail at any time during normal reactor operation. Further, it was assumed that this event would occur in an irradiated fuel assembly with high burnup. This accident was assumed without any consideration of mechanisms that could cause the failure of all cladding. The accident would cause the gaseous fission products and halogens collected in the fuel-clad gap to be released to the reactor pool. The likelihood of such an event was assumed to be 1 in 10,000 years.

The failed fuel assembly was assumed to have been operated at a power density 2.25 times that of the average. It was assumed that the assembly had a burnup of 5,157 megawatt days, which would occur at the end of a 10-year fuel cycle.

The fraction of fission gases released to the fuel-clad gap depends on the operating temperature of the fuel. Based on the calculated fuel temperature for the peak rod of less than 300 °C (572 °F), the fraction of volatile fission products that would escape the fuel material would be 1.5×10^{-5} (Simnad 1980; West, Simnad, and Copeland 1986). For this analysis, the fractional release was conservatively assumed to be 1×10^{-4}, which corresponds to an average operating fuel material temperature of 490 °C (914 °F).

One hundred percent of the noble gases and tritium gas collected in the fuel-clad gap would be released from the fuel assembly and subsequently enter the reactor room. Twenty-five percent of the halogens in the fuel-clad gap would be released from the fuel assembly, and 90 percent of the released halogens would be absorbed in the 9.1-meter-deep (30-foot deep) reactor pool before entering the reactor room. All the radioactive noble gases and halogens that were released to the reactor room are assumed to enter the environment through the reactor building exhaust stack after passing through an activated charcoal filter. The charcoal filter is assumed to remove 99 percent of the halogens (NRC 1978). These assumptions result in an overall release fraction of 1×10^{-4} for the noble gases and tritium gas and 2.5×10^{-8} ($10^{-4} \times 0.25 \times 0.1 \times 0.01$) for the halogens.

A neptunium-237 target assembly is assumed to be damaged along with the fuel assembly. The same release fractions are assumed for the neptunium-237 target as the fuel.

The release to the environment is assumed to occur over 1 hour without decay. This assumption is conservative, because the concentration of the fission products in the reactor room would activate the emergency ventilation system, thereby reducing the room air exchange rate and extending release duration, thus resulting in further decay of the short-lived isotopes.

The radioactive noble, tritium, and halogen gases that would be released to the environment from the design-basis accident scenario are provided in **Table I–29**.

I.1.1.7.2 Fuel- and Target-Handling Accidents

Fuel movements would occur once every 10 years when the whole core (68 fuel assemblies) would be replaced with fresh fuel assemblies. Neptunium-237 target movements would occur once a year. Each year, the irradiated target rods would be removed from the fuel assemblies, packaged in cans, and transferred to the

Table I–29 Design-Basis Accident Source Term

Isotope	Fuel Assembly Inventory (curies)	Neptunium-237 Target Inventory per Gram of Plutonium-238 (curies)	Neptunium-237 Target Assembly Inventory[a] (curies)	Environmental Release (curies)
Hydrogen-3	60.8	0.00241	0.266	0.00611
Krypton-83m	6,530	2.86	316	0.685
Krypton-85	1,440	0.0202	2.23	0.144
Krypton-85m	1.43×10^{4}	5.30	585	1.49
Krypton-87	2.90×10^{4}	8.83	975	3.00
Krypton-88	4.08×10^{4}	0.124	1,370	4.22
Xenon-131m	554	0.303	33.5	0.0587
Xenon-133	9.30×10^{4}	61.3	6,770	9.98
Xenon-133m	2,810	2.14	236	0.305
Xenon-135	5.16×10^{4}	7.69	849	5.24
Xenon-135m	1.76×10^{4}	15.4	1,700	1.93
Xenon-138	8.40×10^{4}	46.7	5,160	8.92
Bromine-82	122	0.0422	4.66	3.17×10^{-6}
Bromine-83	6,480	2.86	316	1.70×10^{-4}
Bromine-84	1.23×10^{4}	4.24	468	3.19×10^{-4}
Iodine-128	278	0.0782	8.63	7.17×10^{-6}
Iodine-130	247	0.273	30.1	6.93×10^{-6}
Iodine-131	4.22×10^{4}	32.5	3,590	0.00114
Iodine-132	6.22×10^{4}	48.7	5,380	0.00169
Iodine-133	9.27×10^{4}	65.0	7,180	0.00250
Iodine-134	1.05×10^{5}	69.0	7,620	0.00282
Iodine-135	8.76×10^{4}	60.8	6,710	0.00236

a. Contains 110.4 grams of plutonium-238 (four target rods of 27.6 grams of plutonium-238).
Source: Calculated results.

spent fuel pool for temporary cooling and storage. The medical and industrial isotope movements would occur more frequently depending on the isotope. The likelihood of a fuel assembly or target drop is estimated to be in the range of 0.01 to 0.0001 per year, or an unlikely event. For this analysis, the likelihood is estimated to be 0.01 per year.

The drop of a fuel assembly could lead to releases of radioactive fission gases. Since the fuel rods are protected by the assembly shroud, fuel damage would be minimal. It is assumed that the drop would damage one fuel rod, releasing the gaseous fission products and halogens to the reactor pool. It is also assumed that the earliest fuel movement would start about 24 hours after the reactor was shut down. Since handling activities would be performed under 3 meters (10 feet) of water, the halogens and gaseous fission products release fractions are assumed to be the same as those for the design-basis accident. The estimated radioactive material release from this accident is provided in **Table I–30**.

A neptunium-237 target assembly consists of four target rods, each containing approximately 27.6 grams (0.97 ounces) of plutonium-238. As these rods are not protected, a drop could lead to a breach of all four.

The target rods are made from neptunium oxides. The fission gas release fraction from the target material to the gap would be similar to that from uranium oxides. Fractional fission gas release was estimated using American National Standards Institute 5.4 (ANSI 1982) and the low-temperature release calculation method. Target rod temperature is not expected to be greater than that of the cladding temperature of an average-power-density fuel rod (approximately 80 °C [176 °F]). For an estimated target rod burnup of

Table I–30 Fuel-Handling Accident Source Term

Isotope	Fuel Rod Inventory (curies)	Environmental Release
Hydrogen-3	0.950	9.50×10^{-5}
Krypton-83m	0.413	4.13×10^{-5}
Krypton-85	22.5	0.00225
Krypton-85m	550	0.0550
Krypton-87	9.55×10^{-4}	9.55×10^{-8}
Krypton-88	1.81	1.81×10^{-4}
Xenon-131m	8.56	8.56×10^{-4}
Xenon-133	1,400	0.140
Xenon-133m	39.8	0.00398
Xenon-135	420	0.0420
Xenon-135m	17.8	0.00178
Bromine-82	1.19	2.98×10^{-8}
Bromine-83	0.108	2.70×10^{-9}
Iodine-130	3.06	7.65×10^{-8}
Iodine-131	614	1.54×10^{-5}
Iodine-132	802	2.01×10^{-5}
Iodine-133	670	1.68×10^{-5}
Iodine-134	4.00×10^{-5}	1.00×10^{-12}
Iodine-135	109	2.73×10^{-6}

706 megawatt days per metric ton of heavy metal, about 0.01 percent of both the long- and short-lived noble gases and halogen gases would be available for release.

As in the fuel-handling accident, 100 percent of the noble and tritium gases in the fuel-clad gap would be released to the environment through the reactor building exhaust system. This results in an overall release fraction of 1×10^{-4} for the noble gases and tritium. Twenty-five percent of the iodine in the fuel-clad gap would be released from the fuel assembly, and 90 percent of the released iodine would be absorbed in the reactor pool. The remaining iodine would be released to the environment through the Reactor Building exhaust system. The exhaust system charcoal filter is assumed to remove 99 percent of the iodine (NRC 1978). This results in an overall release fraction of 2.5×10^{-8} ($10^{-4} \times 0.25 \times 0.1 \times 0.01$) for the iodine.

These assumptions result in the source term shown in **Table I–31**.

Medical, industrial, and research and development isotope targets could also be damaged from a drop accident. Only targets which produce noble gases and halogens either as products or byproducts (including decay) need be considered for analysis. Since the fuel-handling activities are performed under 3 meters (10 feet) of water, these will be the isotopes that have releases to the environment. The iodine-125 product target consequences bound those of the other possible medical and industrial isotope targets.

The iodine-125 product target is assumed to release 100 percent of its inventory to the water. Interaction with the water removes 90 percent of the iodine. The building exhaust system charcoal filters then removes 99 percent of the iodine released from the water. This results in a release fraction of 0.001. The iodine-125 product target would contain approximately 2,530 curies of iodine-125. The estimated radioactive material release from this accident is 2.53 curies of iodine-125.

Table I–31 Neptunium-237 Target-Handling Accident Source Term

Isotope	Neptunium-237 Target Inventory per Gram of Plutonium-238 (curies)	Neptunium-237 Target Assembly Inventory[a] (curies)	Environmental Release (curies)
Hydrogen-3	0.00241	0.266	2.66×10^{-5}
Krypton-83m	2.86	316	0.0316
Krypton-85	0.0202	2.23	2.23×10^{-4}
Krypton-85m	5.30	585	0.0585
Krypton-87	8.83	975	0.0975
Krypton-88	12.4	1,370	0.137
Xenon-131m	0.303	33.5	0.00335
Xenon-133	61.3	6,770	0.677
Xenon-133m	2.14	236	0.0236
Xenon-135	7.69	849	0.0849
Xenon-135m	15.4	1,700	0.170
Xenon-138	46.7	5,160	0.516
Bromine-82	0.0422	4.66	1.16×10^{-7}
Bromine-83	2.86	316	7.89×10^{-6}
Bromine-84	4.24	468	1.17×10^{-5}
Iodine-128	0.0782	8.63	2.16×10^{-7}
Iodine-130	0.273	30.1	7.53×10^{-7}
Iodine-131	32.5	3,590	8.97×10^{-5}
Iodine-132	48.7	5,380	1.34×10^{-4}
Iodine-133	65.0	7,180	1.79×10^{-4}
Iodine-134	69.0	7,620	1.90×10^{-4}
Iodine-135	60.8	6,710	1.68×10^{-4}

a. Contains 110.4 grams of plutonium-238 (four target rods of 27.6 grams of plutonium-238).

I.1.1.7.3 Beyond-Design-Basis Accident

The beyond-design-basis accident assumes an earthquake with sufficient energy to cause structural and equipment failure. The likelihood of such an event was assumed to be the equal to the Reactor Building performance goal for a Performance Category 4 structure. The performance goal for the Reactor Building is 1×10^{-5}, a safety factor of 10 over the return period of 1 in 10,000 years for a Performance Category 4 structure (DOE 1994b). Performance Category 4 is the highest deterministic seismic design criteria for structures, systems, and components in accordance with DOE standards (DOE 1993, 1994b). A performance goal of 1×10^{-5} refers to the annual probability that a seismic event would cause damage to a component so that it could not perform its function. Therefore, an earthquake level with a return period of 1 in 100,000 years was assumed to initiate the beyond-design-basis accident. Since both the reactor pool and the spent fuel pool would be designed to withstand a higher-level earthquake than that for Performance Category 4, no failure of these pools was assumed. However, it was assumed that the equipment and systems that support these pools would fail. Further, it is assumed that the earthquake would initiate reactor scram (loss of power would cause the control rods to drop in the reactor core), damage the cooling pipe outside of the reactor pool, and possibly breach the reactor room confinement.

Since the accident would not result in a loss of reactor pool coolant below the level at which primary piping leaves the pool and the reactor shuts down, sufficient coolant would be available to keep the core covered for

at least 40 days after the accident. For this analysis, it was assumed that the fuel-handling crane above the pool would fall into the pool and damage the core tank and fuel rods inside the core. This assumption is conservative since the top of the core tank would be 0.61 meters (2 feet) above the top of the fuel assemblies acting as a chimney to enhance natural-convection core cooling during reactor shutdowns. In addition, the top of the active fuel is another foot below the top of the fuel assembly. Therefore, the crane would have to damage both the upper core barrel and the top of the fuel assemblies before it could damage the fuel. Nevertheless, it was assumed that the drop would cause fuel damage.

The drop was assumed to cause releases of all gaseous fission products and halogens through the pool water directly to the environment, bypassing the charcoal filter and the building exhaust stack. For the fuel and neptunium-237 targets, the release fractions are 1×10^{-4} for the noble gases and tritium and 2.5×10^{-6} for the halogens. For the medical isotope targets, the release fractions are 1.0 for noble gases and 0.1 for halogens.

The new research reactor core consists of 68 fuel assemblies with a total of 4,080 fuel rods; 48 neptunium-237 assemblies with 4 target rods each; 8 medical, industrial, and research and development target assemblies with 2 target rods each; and 8 rabbit tubes. For this analysis, the 8 medical, industrial, and research and development target assemblies are assumed to contain the xenon-127 product target. The rabbit tubes would contain 7 iodine-131 product targets and 1 iodine-125 product target. This core configuration results in the highest consequences from accidental releases.

This core configuration and these release fraction assumptions result in the source term presented in **Table I–32**.

Table I–32 Beyond-Design-Basis Earthquake Accident Source Term

Isotope	Fuel Core Inventory[a] (curies)	Medical, Industrial, and Research and Development Isotope Core Inventory[b] (curies)	Neptunium-237 Target Inventory per Gram of Plutonium-238[c] (curies)	Neptunium-237 Target Core Inventory[d] (curies)	Environmental Release (curies)
Hydrogen-3	1,790		0.00241	12.8	0.180
Krypton-85	4.60×10^{4}		0.0202	107	4.61
Krypton-85m	5.30×10^{5}		5.30	2.81×10^{4}	55.8
Krypton-87	7.27×10^{5}		8.83	4.68×10^{4}	77.4
Krypton-88	1.38×10^{6}		12.4	6.57×10^{4}	145
Iodine-125		2,530	0.0	0.0	253
Iodine-131	1.59×10^{6}	2,150	32.5	1.72×10^{5}	219
Iodine-132	2.35×10^{6}		48.7	2.58×10^{5}	6.52
Iodine-133	3.53×10^{6}		65.0	3.44×10^{5}	9.69
Iodine-134	3.03×10^{6}		69.0	3.66×10^{5}	8.49
Iodine-135	3.04×10^{6}		60.8	3.22×10^{5}	8.41
Xenon-127		116			116
Xenon-133	3.60×10^{6}		61.3	3.05×10^{5}	392
Xenon-135	2.72×10^{6}		7.69	4.07×10^{4}	276

a. Fuel inventory lists only those isotopes with an environmental release.
b. Medical, industrial, and research and development inventory lists only the isotopes with the highest environmental release.
c. Neptunium-237 inventory lists only those isotopes with an environmental release.
d. Based on a 5-kilogram-per-year plutonium-238 production rate.

I.1.1.7.4 Decontamination and Decommissioning Accidents

The decontamination and decommissioning activities would be performed according to a preestablished plan, known as the decommissioning plan. Activities would include decontamination and dismantling of reactor components, removal of spent nuclear fuel, cleaning and removal of the reactor pool and spent fuel pool water, decontamination and dismantling of equipment and structures, and preparation of the site for unrestricted use. These activities could potentially result in an accidental release of radioactive material. Radioactive releases could occur from improper cutting of activated components and equipment, dropping of a radioactively contaminated component, and from spills of contaminated liquids. The potential on- and offsite impacts of accidents would be expected to be less than, or within, the values estimated for occurrences during normal operations.

At this preconceptual research reactor design stage, the major areas with the greatest inventory of radioactivity would be the spent fuel pool and the components within the reactor pool and primary coolant system. The spent fuel pool would contain about 272 spent fuel assemblies, that is, four full core loads. The minimum decay times for each assembly would range between 5 and 30 years. The assumption is that spent fuel removal would begin 5 years after the last core was removed from the reactor. Once the fuel assemblies were removed from the core, the beryllium reflector and the reactor core tank would contain the highest radioactive inventory of tritium and cobalt-60 in the reactor pool area.

A spectrum of accidents was evaluated considering activities that would occur during decontamination and decommissioning of the research reactor and support facility. It was determined that two accidents had the greatest potential for onsite and offsite impacts: a drop of a spent nuclear fuel cask during fuel removal and an accidental vaporization of a small segment of the reactor core tank during dismantlement.

SPENT FUEL CASK DROP ACCIDENT

The lifting capability of the spent fuel pool crane would be limited to truck-sized spent nuclear fuel transportation casks that would be used to move the spent fuel to a central storage location. The cask would be loaded under water, the cask cover would be installed but not tightly sealed, and the cask raised above the water where it would be sprayed with demineralized water before it was put on the ground for decontamination and draining of pool water. The cask then would be sealed, backfilled with inert gas, and moved to be loaded onto the truck trailer bed. The maximum lift would be less than 9.1 meters (30 feet) above the pool floor, or less than 30 centimeters (1 foot) above the spent fuel pool building floor level.

A spent nuclear fuel cask was assumed to drop while it was stopped to be rinsed. The drop would not damage the cask or the spent fuel pool liner. The cask is designed to withstand a drop from 9.1 meters (30 feet) onto an unyielding surface without failure. The cask would not be lifted above 9.1 meters (30 feet) above the ground, and the drop over the spent fuel pool would hit the pool surface which provides 7.62 meters (25 feet) of water acting as a damper, reducing the impact velocity. Therefore, no damage to the spent fuel pool liner would be expected.

The fuel rods in the cask would be protected from damage not only by the cask, but also by the assembly shroud. However, for this analysis, it was assumed that one row of fuel in one assembly would fail and release the gaseous fission products from the fuel-clad gap. The fraction of fission gases released to the fuel-clad gap was conservatively assumed to be 1×10^{-4}. One hundred percent of the noble gases and tritium gas in the fuel-clad gap would be released through the pool to the reactor room. Twenty-five percent of the halogens in the fuel-clad gap would be released and 90 percent of the released halogens would be absorbed in the reactor pool before entering the reactor room. All the noble gases, tritium, and halogens that enter the reactor room would be released to the environment through the reactor building exhaust system after passing through an

activated charcoal filter. The charcoal filter was assumed to remove 99 percent of the halogens (NRC 1978). These assumptions result in a release fraction of 1×10^{-4} for noble gases and tritium and 2.5×10^{-8} for halogens.

The likelihood of such an accident was estimated to be less than 5 in 1 million, or 5×10^{-6} per year. This estimate was derived from a recent NRC technical study of spent fuel accident risk at decommissioning nuclear power plants (NRC 2000). Based on an assumption of 100 heavy-load cask lifts per year, the NRC estimated a cask drop mean frequency of 9.6×10^{-6} per year. Considering that the total number of spent fuel cask lifts at this facility would be less than 40, assuming that all the fuel would be shipped offsite in a year, the cask drop frequency would be less than 5×10^{-6} per year for that year.

For analysis, the frequency of this accident was assumed to be 5.0×10^{-6} per year. The source term for the spent fuel cask drop accident is presented in **Table I–33**.

Table I–33 Spent Fuel Cask Drop Accident Source Term

Isotopes	Fuel Assembly - Fission Gas and Halogen Inventory (curies)	Eight Fuel Rods - Fission Gas and Halogen Inventory (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3 (tritium)	2.12×10^{1}	2.65	1.00×10^{-4}	2.65×10^{-4}
Krypton-85	1.44×10^{3}	1.80×10^{2}	1.00×10^{-4}	1.80×10^{-2}
Iodine-129	1.86×10^{-3}	2.33×10^{-4}	2.50×10^{-8}	5.81×10^{-12}

REACTOR CORE TANK VAPORIZATION ACCIDENT

An accidental vaporization of a small segment of the reactor core tank during size reduction was assumed. The stainless steel reactor core tank would need to be cut into pieces in order to be transported offsite. The major activation product in the tank would be cobalt-60, with an inventory of 359 curies after a 5-year decay time. Plasma torches would most likely be used for the process. The cutting process would occur with strict radiological controls under a tent with proper ventilation to collect any vaporized particulates. The vaporized particulates would be passed through HEPA filters before exhausting to the environment.

For this analysis, it was assumed that the exhaust system would fail and that the torch would not shut down and would vaporize a small segment of the tank. It was assumed that the torch would burn through the wall of the tank creating a 6.25-square centimeter (1-square inch) hole in the wall. The frequency of this accident was assumed to be 1×10^{-4}. The source term for the reactor core tank vaporization accident is estimated to be 0.026 curies of cobalt-60 released directly to the environment.

I.1.2 Methodology for Estimating Irradiation Facility Accident Radiological Impacts

The MACCS2 computer code (Version 1.12) was used to estimate the consequences of the postulated accidents. A detailed description of the MACCS model is provided in NUREG/CR-4691 (Chanin et al. 1990). The enhancements incorporated in MACCS2 are described in the MACCS2 User's Guide (Chanin and Young 1997). Originally developed to model the radiological consequences of nuclear reactor accidents, this code has been used for the analysis of accidents for many EISs and other safety documentation, and is considered applicable to the analysis of accidents associated with the production of plutonium-238 and other proposed isotopes.

MACCS2 models the offsite consequences of an accident that releases a plume of radioactive materials to the atmosphere. Should such an accidental release occur, the radioactive gases and aerosols in the plume would be transported by the prevailing wind and dispersed into the atmosphere, and the population would be exposed

to radiation. The atmospheric dispersion is modeled on a polar-coordinate spatial grid centered on the facility and extending out to 80 kilometers (50 miles). The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid correspond to the 16 directions of the compass. MACCS2 generates the distribution of downwind doses at specified distances, as well as the distribution of population doses.

Radiological consequences may vary somewhat as a result of variations in the duration of release. For longer releases, there is a greater chance of plume meander (i.e., changes in flow attributable to variations in wind direction over the duration of release). MACCS2 models plume meander by increasing the lateral dispersion coefficient of the plume for longer release durations, thus lowering the dose. The other effect of longer release durations is involvement of a greater variety of meteorological conditions in a given release, which reduces the variance of the resulting dose distributions. This would tend to lower high-percentile doses, raise low-percentile doses, and have no effect on the mean dose.

The MACCS2 code was applied in a probabilistic manner using a weather bin–sampling technique. The weather bin–sampling method sorts weather sequences into categories and assigns a probability to each category according to the initial conditions (wind speed and stability class) and the occurrence of rain. Each of the sampled meteorological sequences was applied to each of the 16 sectors accounting for the frequency of occurrence of the wind blowing in that direction (i.e., site compass sector wind rose frequencies). Individual doses, as a function of distance and direction, were calculated for each of the meteorological sequence samples. The mean dose values of the sequences were generated for each of the 16 sectors. The highest of these dose values was used as the dose delivered to the maximally exposed offsite individual and the noninvolved worker. Population doses within 80 kilometers (50 miles) of each facility were also calculated.

In addition to short-term health effects of exposure to the plume passage, long-term effects were also modeled. The long-term health effects include direct exposure to contaminated ground and inhalation of resuspended materials, as well as indirect health effects of the consumption of contaminated food and water. Long-term protective measures such as decontamination, temporary relocation, contaminated crops, milk condemnation, and prohibition of farmland production are based on EPA Protection Action Guides.

For each potential accident, information is provided on accident consequences and frequencies to three types of receptors: (1) a noninvolved worker, (2) the maximally exposed individual, and (3) the offsite population. The first receptor, a noninvolved worker, is a hypothetical individual working on site but not involved in the proposed activity. The worker is assumed to be downwind at a point 640 meters (0.4 miles) from the accident. Although other distances closer to the accident could have been assumed, the calculations break down at distances of about 200 meters (656 feet) or less due to limitations in modeling of the effects of building wake and local terrain on dispersion of the released radioactive substances. A worker closer than 640 meters (2,100 feet) to the accident would generally receive a higher dose; a worker farther away, a lower dose. The second receptor, the maximally exposed individual, is a hypothetical individual assumed to be downwind at the site boundary or on a highway within the site boundary, whichever results in a higher dose. Exposures received by this individual are intended to represent the highest doses to a member of the public. The third receptor, the offsite population, is all members of the public within 80 kilometers (50 miles) of the accident location.

It is possible that an individual member of the public could be closer to a facility than either the site boundary or the nearest onsite highway. However, such individuals would be present only occasionally and for brief periods (a few hours or more). Hence, the annual probability that an individual would be close is relatively low, and the associated risk to that individual would be bounded by the site boundary or onsite highway maximally exposed individual risk.

For the CLWR analysis, a noninvolved worker was not evaluated for two reasons. First, the noninvolved worker was originally developed for large DOE sites, where several different facilities are under facility-specific control. The noninvolved worker is an individual not under specific facility control, but also not outside the site boundary. At a CLWR, however, the entire site is within the exclusion area and under the same control.

Second, each accident scenario has a warning time and a subsequent release time. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is the time when the release to the environment begins. The minimum time between the warning time and the release time for this analysis is 20 minutes. Twenty minutes is enough time to evacuate onsite personnel. It is also conservatively assumed that an onsite emergency has not been declared prior to initiating offsite notification.

Consequences to involved workers are addressed in Section I.1.7.

All radiological impacts are calculated in terms of committed dose and associated health effects for an individual or exposed population. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent from the external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of rem for individuals and person-rem for a population. The impacts are further expressed as health risks, specifically in terms of latent cancer fatalities.

The health risks for a noninvolved worker and the maximally exposed offsite individual are expressed as the additional potential or likelihood of a latent cancer fatality. The health risk to the population is expressed as the increased number of latent cancer fatalities.

The probability coefficients for determining the likelihood of latent cancer fatality, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991). For low doses or dose rates, respective probability coefficients of 0.0004 and 0.0005 fatal cancer per rem are applied for workers and the general public. For high doses received at a high rate, respective probability coefficients of 0.0008 and 0.001 fatal cancer per rem are applied for noninvolved workers and the public. These higher probability coefficients apply where doses are above 20 rad or dose rates are above 10 rad per hour.

Tritium releases were modeled as tritiated water vapor rather than elemental tritium. Tritiated water is more effectively absorbed by humans and therefore results in a much greater health hazard.

I.1.2.1 Uncertainties

The analyses of accidents are based on calculations relevant to hypothetical sequences of events and models of their effects. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and the effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, the scarcity of experience with the accidents postulated leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives.

Of particular interest are the uncertainties in the estimate of cancer deaths from exposure to radioactive materials. The numerical values of the health risk estimates used in this NI PEIS were obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality resulting from

exposures at 10 rad. Other methods of extrapolation to the low-dose region could yield higher or lower estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation, and the possibility of no risk or even health benefits (hormesis effects) cannot be excluded. Because the health risk estimators are multiplied by conservatively calculated radiological doses to predict fatal cancer risks, the fatal cancer values presented in this NI PEIS are expected to be overestimates.

For the purposes of presentation in this NI PEIS, the impacts calculated from the linear model are treated as an upper-bound case, consistent with the widely used methodologies for quantifying radiogenic health impacts. This does not imply that health effects are expected. Moreover, in cases where the upper-bound estimators predict a number of latent cancer deaths that is greater than 1, this does not imply that the latent cancer death can be determined for a specific individual.

I.1.3 Irradiation Facility Accident Consequences and Risks

The irradiation facility accident source term data presented in Sections I.1.1.2–I.1.1.7 were evaluated using the MACCS2 accident analysis computer code. **Tables I–34** through **I–41** summarize the consequences and risks of the accidents, with mean meteorological conditions, for the maximally exposed individual, the offsite population within an 80-kilometer (50-mile) radius of the facility, and a noninvolved worker 640 meters (2,100 feet) from the release point. As explained in Section I.1.2, noninvolved worker consequences were not evaluated for the CLWR accidents.

Table I–34 presents ATR accident consequences and risks for three possible plutonium-238 production rates: 0, 3, and 5 kilograms (0, 6.6, and 11 pounds) per year.

Table I–35 presents HFIR accident consequences and risks for two possible plutonium-238 production rates: 0 and 2 kilograms (0 and 4.4 pounds) per year.

Table I–36 presents CLWR accident consequences and risks for two possible plutonium-238 production rates: 0 and 5 kilograms (0 and11 pounds) per year.

Severe-accident scenarios that postulate large, abrupt releases could result in early fatalities if the radiation dose were sufficiently high. For the irradiation facilities analyzed, early fatalities are postulated only for the early containment failure and containment bypass event at the generic CLWR.

Table I–37 presents CLWR-estimated early fatalities and associated risks for two possible plutonium-238 production rates: 0 and 5 kilograms (0 and 11 pounds) per year.

Table I–38 presents FFTF accident consequences and risks for simultaneous medical, industrial, research and development, and plutonium-238 production for both mixed oxide and highly enriched uranium fuels.

Table I–39 presents accelerator accident consequences and risks for medical, industrial, research and development, and plutonium-238 isotope production.

Table I–40 presents new research reactor accident consequences and risks for the simultaneous medical, industrial, research and development, and plutonium-238 isotope production.

Table I–41 presents new research reactor decontamination and decommissioning accident consequences and risks.

Table I–34 ATR Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	0.465	2.33×10^{-4}	2.33×10^{-8}	5.11×10^{4}	25.5	0.00255	5.15	0.00206	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production (1×10^{-4})	0.549	2.75×10^{-4}	2.75×10^{-8}	5.15×10^{4}	25.7	0.00257	6.52	0.00261	2.61×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production (1×10^{-4})	0.604	3.02×10^{-4}	3.02×10^{-8}	5.17×10^{4}	25.9	0.00259	7.61	0.00304	3.04×10^{-7}
Target-handling with 0 kg/yr plutonium-238 production (0.001)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Target-handling with 3 kg/yr plutonium-238 production (0.001)	1.23×10^{-4}	6.15×10^{-8}	6.15×10^{-11}	0.0786	3.93×10^{-5}	3.93×10^{-8}	0.00195	7.80×10^{-7}	7.80×10^{-10}
Target-handling with 5 kg/yr plutonium-238 production (0.001)	2.05×10^{-4}	1.03×10^{-7}	1.03×10^{-10}	0.128	6.41×10^{-5}	6.41×10^{-8}	0.00324	1.30×10^{-6}	1.30×10^{-9}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: ATR, Advanced Test Reactor; kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–35 HFIR Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.41	0.00121	1.21×10^{-7}	2,990	1.49	1.49×10^{-4}	17.2	0.00688	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production (1×10^{-4})	2.41	0.00121	1.21×10^{-7}	3,000	1.50	1.50×10^{-4}	17.2	0.00688	6.88×10^{-7}
Target-handling with 0 kg/yr plutonium-238 production (0.001)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Target-handling with 2 kg/yr plutonium-238 production (0.001)	4.96×10^{-4}	2.48×10^{-7}	2.48×10^{-10}	0.335	1.68×10^{-4}	1.68×10^{-7}	0.00245	9.80×10^{-7}	9.80×10^{-10}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: HFIR, High Flux Isotope Reactor; kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–36 Generic CLWR Accident Consequences and Risks

Accident (Frequency)	Annual Plutonium-238 Production (kilograms per year)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)		
		Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]
Containment early failure (7.92×10^{-8})	0	3350	1.00[e]	7.92×10^{-8}[f]	1.80×10^{6}	1,250	9.89×10^{-5}
	5	3670	1.00[e]	7.92×10^{-8}[f]	1.90×10^{6}	1,340	1.06×10^{-4}
Containment late failure (1.07×10^{-5})	0	1.11	5.55×10^{-4}	5.94×10^{-9}	1.06×10^{5}	53.6	5.74×10^{-4}
	5	1.12	5.60×10^{-4}	5.99×10^{-9}	1.06×10^{5}	53.6	5.74×10^{-4}
LOCA (4.65×10^{-5})	0	0.0312	1.56×10^{-5}	7.25×10^{-10}	186	0.0931	4.33×10^{-6}
	5	0.0313	1.57×10^{-5}	7.30×10^{-10}	187	0.0935	4.35×10^{-6}
Containment bypass (1.53×10^{-6})	0	1540	1.00[e]	1.53×10^{-6}[f]	1.45×10^{6}	922	1.41×10^{-3}
	5	1680	1.00[e]	1.53×10^{-6}[f]	1.52×10^{6}	978	1.49×10^{-3}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred. MACCS2 calculates the dose to each exposed individual in the population, applies the appropriate cancer risk factor, and then sums the individual probabilities to determine the number of latent cancer fatalities.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: CLWR, commercial light water reactor; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–37 Generic CLWR Early Fatality Consequences and Risks

Accident (Frequency)	Reactor Annual Plutonium-238 Production (kilograms per year)	Population to 80 Kilometers (50 Miles)	
		Early Fatalities[a]	Annual Risk[b]
Early containment failure (7.92×10^{-8})	0	8.65	6.85×10^{-7}
	5	8.76	6.94×10^{-7}
Containment bypass (1.53×10^{-6})	0	3.48	5.32×10^{-6}
	5	3.51	5.37×10^{-6}

a. Number of early fatalities assuming the accident occurred.
b. Increased number of early fatalities per year.
Key: CLWR, commercial light water reactor.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–38 FFTF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	LCF[a]	Annual Risk[b]	Dose (person-rem)	LCF[c]	Annual Risk[d]	Dose (rem)	LCF[a]	Annual Risk[b]
Design-basis accident primary sodium spill (MOX) (1×10^{-4})	0.00113	5.65×10^{-7}	5.65×10^{-11}	78.6	0.0393	3.93×10^{-6}	0.00313	1.25×10^{-6}	1.25×10^{-10}
Design-basis accident primary sodium spill (HEU) (1×10^{-4})	8.63×10^{-4}	4.32×10^{-7}	4.32×10^{-11}	72.6	0.0363	3.63×10^{-6}	0.00181	7.24×10^{-7}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})[e]	0.679	3.40×10^{-4}	3.40×10^{-10}	6.68×10^{4}	33.4	3.34×10^{-5}	0.679	2.72×10^{-4}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})[e]	0.481	2.41×10^{-4}	2.41×10^{-10}	6.16×10^{4}	30.8	3.08×10^{-5}	0.375	1.50×10^{-4}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	0.00383	1.92×10^{-6}	1.92×10^{-13}	1,280	0.639	6.39×10^{-8}	0.357	1.43×10^{-4}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	0.00384	1.92×10^{-6}	1.92×10^{-13}	1,230	0.617	6.17×10^{-8}	0.340	1.36×10^{-4}	1.36×10^{-11}
BLTC plutonium-238 target-handling accident (1×10^{-7})	2.61×10^{-4}	1.31×10^{-7}	1.31×10^{-14}	25.8	0.0129	1.29×10^{-9}	0.0279	1.12×10^{-5}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	1.22×10^{-4}	6.10×10^{-8}	6.10×10^{-15}	2.74	0.00137	1.37×10^{-10}	0.0143	5.72×10^{-6}	5.72×10^{-13}
Standby accident (1×10^{-4})	1.34×10^{-7}	6.70×10^{-11}	6.70×10^{-15}	0.00999	4.99×10^{-6}	4.99×10^{-10}	1.62×10^{-8}	6.48×10^{-12}	6.48×10^{-16}
Deactivation accident (0.10)[f]	4.75×10^{-10}	2.38×10^{-13}	2.38×10^{-14}	3.64×10^{-5}	1.82×10^{-8}	1.82×10^{-9}	3.88×10^{-9}	1.55×10^{-12}	1.55×10^{-13}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Frequency incorporates external initiators.
f. Frequency is per event (deactivation) rather than per year.

Key: BLTC, bottom-loading transfer cask; FFTF, Fast Flux Test Facility; HEU, highly enriched uranium fuel; LCF, latent cancer fatalities; MOX, mixed oxide fuel.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–39 Accelerator Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
High-energy accelerator									
Design-basis target-handling accident (1×10^{-4})	2.93×10^{-4}	1.47×10^{-7}	1.47×10^{-11}	9.80×10^{-1}	4.90×10^{-4}	4.90×10^{-8}	9.35×10^{-4}	3.74×10^{-7}	3.74×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	11.7	5.85×10^{-3}	5.85×10^{-8}	3.01×10^{4}	18.0	1.80×10^{-4}	184	1.47×10^{-1}	1.47×10^{-6}
Low-energy accelerator									
Design-basis target-handling accident (1×10^{-4})	8.05×10^{-5}	4.03×10^{-8}	4.03×10^{-12}	17.7	8.85×10^{-3}	8.85×10^{-7}	1.12×10^{-3}	4.48×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	1.32×10^{-2}	6.60×10^{-6}	6.60×10^{-11}	32.4	1.62×10^{-2}	1.62×10^{-7}	2.08×10^{-1}	8.32×10^{-5}	8.32×10^{-10}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results using MACCS2 (Chanin and Young 1997).

Table I–40 New Research Reactor Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Design-basis accident (1×10^{-4})	1.33×10^{-6}	6.65×10^{-10}	6.65×10^{-14}	2.41×10^{-3}	1.20×10^{-6}	1.20×10^{-10}	5.49×10^{-6}	2.20×10^{-9}	2.20×10^{-13}
Beyond-design-basis earthquake (1×10^{-5})	0.00373	1.87×10^{-6}	1.87×10^{-11}	27.6	1.38×10^{-2}	1.38×10^{-7}	0.0531	2.12×10^{-5}	2.12×10^{-10}
Fuel-handling accident (0.01)	1.90×10^{-9}	9.50×10^{-13}	9.50×10^{-15}	6.79×10^{-6}	3.40×10^{-9}	3.40×10^{-11}	5.83×10^{-9}	2.33×10^{-12}	2.33×10^{-14}
Neptunium-237 target-handling accident (0.01)	5.42×10^{-8}	2.71×10^{-11}	2.71×10^{-13}	8.95×10^{-5}	4.47×10^{-8}	4.47×10^{-10}	2.43×10^{-7}	9.72×10^{-11}	9.72×10^{-13}
Medical isotope target-handling accident (0.01)	1.04×10^{-5}	5.20×10^{-9}	5.20×10^{-11}	0.101	5.06×10^{-5}	5.06×10^{-7}	6.76×10^{-6}	2.70×10^{-9}	2.70×10^{-11}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results, using the GENII (Napier et al. 1988) and MACCS2 (Chanin and Young 1997) computer codes.

Table I–41 New Research Reactor Decontamination and Decommissioning Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Spent fuel cask drop (5.0×10^{-6})	7.01×10^{-12}	3.51×10^{-15}	1.75×10^{-20}	2.78×10^{-8}	1.39×10^{-11}	6.95×10^{-17}	1.30×10^{-11}	5.20×10^{-15}	2.60×10^{-20}
Reactor core tank vaporization (1.0×10^{-4})[e]	1.55×10^{-5}	7.75×10^{-9}	7.75×10^{-13}	3.46×10^{-1}	1.73×10^{-4}	1.73×10^{-8}	5.23×10^{-5}	2.09×10^{-8}	2.09×10^{-12}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Frequency per event.

Source: Model results using MACCS2 (Chanin and Young 1997).

I.1.4 Processing Facility Accident Scenario Selection and Description

I.1.4.1 Plutonium-238 Processing

For the processing facilities, a spectrum of accidents was developed that considered a full range of accidents associated with such facilities. The scenarios evaluated, however, represent bounding cases that are considered to envelop the risk profile.

The processing facility accidents presented in the ORNL Radiochemical Engineering Development Center (REDC) Safety Analysis Report for Building 7920 (ORNL 1999) were reviewed for evaluation in this NI PEIS. Process and facility details were based on the preconceptual design study to support plutonium-238 production (Wham et al. 1998). Since process details at the Fluorinel Dissolution Processing Facility (FDPF) and the Fuel and Materials Examination Facility (FMEF) are essentially the same as those at REDC, the same spectrum of accidents was evaluated for all the processing facilities. However, facility differences were accounted for in evaluating the consequences of these accidents.

Several design-basis accidents were selected for inclusion in this NI PEIS. These include:

1. A postulated explosion in a glovebox during neptunium-237 target fabrication, representing the glovebox-handling accident having the largest potential consequences

2. A postulated failure of the target dissolver tank containing both neptunium-237 and plutonium-238, representing the accidental spill having the greatest consequences

3. A postulated explosion of an ion exchange column during plutonium-238 purification, which has the potential to release more plutonium-238 than any other processing facility design-basis accident.

A fire in a hot cell was judged to have lower consequences than an explosion, and was not included in this NI PEIS. This is based on an extensive experimental investigation (Hasegawa et al. 1992), which concluded that a fire in a hot cell would not represent a threat to the effectiveness of the facility roughing or HEPA filters and would be self-extinguishing within a short time from lack of oxygen.

Criticality accidents were not evaluated in this NI PEIS because the risk of accidental criticality, given appropriate administrative and process controls, is considered to be very low. Both neptunium-237 and plutonium-238 would be stored in shielded containers in quantities and configurations that preclude criticality. Target preparation and postirradiation processing will be carried out in batches involving quantities well below those at which criticality could occur. As a result, a criticality accident could occur only as a result of a series of gross, deliberate violations of established controls.

The postulated beyond-design-basis processing facility accident selected for use in this NI PEIS is a catastrophic earthquake resulting in a collapse of the nearby stack and failure of the HEPA filter system intended to mitigate the consequences of releases. Two cases of this accident were evaluated. Case 1 assumed that the facility was only being used to store neptunium-237. Case 2 assumed that the facility was an integrated storage, target fabrication, and irradiated-target-processing facility.

The waste stream from the irradiated targets would be processed in the same facilities as the irradiated targets. Accidents occurring during the processing of the waste stream were not evaluated in this NI PEIS because their consequences are bounded by the irradiated target accidents that have been evaluated.

I.1.4.1.1 Design-Basis Accidents

ION EXCHANGE EXPLOSION DURING NEPTUNIUM-237 TARGET FABRICATION

An accident can occur during fabrication of the neptunium-237 targets. As part of the target preparation, 1-kilogram (2.2-pound) quantities of neptunium-237 solution are processed (Wham et al. 1998) to yield neptunium in an oxide form for use as a target material. This operation takes place in a shielded glovebox and involves use of an ion exchange column. This accident scenario postulates an explosion of the ion exchange column in the glovebox. Judging from historical occurrences of this type of accident at radiochemical laboratories and processing facilities, the frequency of this event is "unlikely" (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

The glovebox is maintained at a slight negative pressure with respect to that portion of the building outside the hot cells, and is continually exhausted to the atmosphere through roughing filters and HEPA filters, and via a stack.

An explosion is estimated to release essentially all of the neptunium-237 into the glovebox. Additional data to calculate releases were taken from relevant facility data (ORNL 1999; Green 1998, 1999) and other accepted sources (DOE 1994a). Since an explosion involves small quantities of materials, any increase in pressure is expected to be small and is not expected to result in transitory leakage of radioactive material from the glovebox into the operating area.

The glovebox is exhausted through roughing filters and then through two banks of HEPA filters arranged in series outside the building and then to the environs via a stack. Each bank of the HEPA filters is assumed to remove 99.95 percent of all particulates at or above a size of 0.3 micron (Burchsted et al. 1976). (Note: This assumes two HEPA filters are in series and each is 99.95 percent efficient, yielding a 2.5×10^{-7} reduction factor.)

Airborne releases can be divided into respirable (smaller than about 10 microns) and nonrespirable fractions. Nonrespirable airborne particles can cause localized onsite contamination, but they do not contribute significantly to offsite doses for several reasons. For design-basis accidents, the filter efficiency for the larger, nonrespirable particles is greater than that for all particles of the respirable fractions, and significantly greater than the minimum value of 99.95 percent for 0.3-micron particles. For the beyond-design-basis earthquake, where filters are postulated to be ineffective, leakage from the hot cells is at a low rate, allowing for increased deposition and settling of the larger particles prior to release. Even where large, nonrespirable particles are released to the environment, their atmospheric transport is limited and they will "fall out" within a short distance from the release point.

Table I–42 shows the release fractions and source terms for this accident.

Table I–42 Neptunium-237 Target Preparation Accident Source Terms

Analysis Parameters	Units
Neptunium-237 inventory in glovebox	1,000 grams
Neptunium-237 released into glovebox from explosion	1,000 grams
Airborne release fraction times respirable particle fraction	7×10^{-2}
Leak path factor	0.50
Neptunium-237 reaching HEPA filters	35.0 grams
Neptunium-237 released from stack to environs	8.75×10^{-6} grams

Source: Calculated results.

TARGET DISSOLVER TANK FAILURE DURING PLUTONIUM-238 SEPARATION

A hypothetical accident considered was the failure of a tank in which irradiated neptunium-237 targets are to be dissolved. The irradiated neptunium-237 target processing is planned to be carried out in approximately five batches per year. Each batch of irradiated targets is expected to contain approximately 1 kilogram (2.2 pounds) of plutonium-238 and 8 to 10 kilograms (17.6 to 22 pounds) of neptunium-237. A complete failure of the dissolver tank envelops a spectrum of accidental spills involving plutonium-238 in the hot cells. The complete failure of this tank is judged to be unlikely (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

This scenario postulates the sudden, complete failure of the dissolver tank and the spilling of its contents onto the floor of the hot cell. The product of the airborne release fraction and the respirable fraction is the sum of that for a free-fall spill, plus evaporation of a shallow pool and are estimated (DOE 1994a) to be 0.00013. A leak path factor of 0.75, applicable for a hot cell (Green 1998), was used.

The cell is exhausted first to roughing filters, then through two stages of HEPA filters in series, and then to the environs via a stack. (Note: This assumes two HEPA filters are in series, and each is 99.95 percent efficient, yielding a 2.5×10^{-7} reduction factor.)

Table I–43 shows the release fractions and source terms for this accident.

Table I–43 Target Dissolver Tank Failure Source Terms

Analysis Parameters	Neptunium-237	Plutonium-238
Inventory in dissolver tank	9,000 grams	1,000 grams
Spilled onto hot cell floor	9,000 grams	1,000 grams
Airborne release fraction times respirable fraction	0.00013	0.00013
Leak path factor	0.75	0.75
Amount entering HEPA filters	0.88 gram	0.098 gram
Amount released from stack to environs	2.19×10^{-7} gram	2.44×10^{-8}gram

Source: Calculated results.

ION EXCHANGE EXPLOSION DURING PLUTONIUM-238 SEPARATION

A hypothetical accident considered is the postulated explosion of an ion exchange column during plutonium-238 purification in a hot cell. Although plans for plutonium purification call for a solvent extraction process, an alternative method involves the use of an ion exchange process (Wham et al. 1998). In this alternative procedure, 495 grams (1.1 pounds) of plutonium-238 are loaded onto an ion exchange column. This postulated accident scenario involves an explosion of this ion exchange column. Judging from historical occurrences of this type of accident at radiochemical laboratories and processing facilities, the frequency of

this event is unlikely (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

Most of the plutonium will be deposited on the cell walls and floor along with other explosion debris. The fraction of plutonium estimated to be released in airborne form and respirable size particles is 0.07 (DOE 1994a).

The hot cell is maintained at a slight negative pressure with respect to the rest of the building. After effluents are exhausted from the hot cell, they pass first through roughing filters, then through two banks of HEPA filters outside the building. On exiting the HEPA filters, effluents are released to the environs through a stack. At the REDC, the explosion could also result in the generation of a weak shock wave and a momentary pressure increase of up to several pounds per square inch gage in the hot cell (ORNL 1999). This accident would not be expected to generate dynamic pressures sufficient to damage the hot cell confinement structure, but could result in some leakage of radioactive materials into the operating areas of the building due to the brief pressurization of the hot cell cubicle (ORNL 1999). Because of the larger volume of the FDPF and FMEF facilities, the magnitude of a shock wave would be much lower.

For REDC, the shock wave may impact the HEPA filters, possibly degrading their performance. Although the HEPA filters are tested to retain 99.97 percent efficiency, tornado conditions are estimated (DOE 1994a) to reduce their efficiency to approximately 99 percent. This scenario assumes that the efficiency of the first-stage HEPA filters at REDC is partially degraded to 99.5 percent while the second-stage efficiency is 99.95 percent. This yields a reduction factor of 2.5×10^{-6} at REDC. Both HEPA stages are 99.95 percent efficient, yielding a reduction factor of 2.5×10^{-7} at FDPF and FMEF. For FDPF and FMEF, the HEPA filters were assumed not to be degraded, because the magnitude of any shock wave generated would be much less. The release to the environment was conservatively assumed to consist of a single "puff" associated with the immediate explosion.

Table I–44 shows the release fractions and source terms for this accident.

Table I–44 Plutonium-238 Ion Exchange Explosion Accident Source Terms

Analysis Parameters	Units
Plutonium-238 material at risk	495 grams
Plutonium-238 released into Hot Cell E from explosion	495 grams
Airborne release fraction times respirable particle fraction	7×10^{-2}
Leak path factor	0.75
Plutonium-238 reaching HEPA filters	26.0 grams
Plutonium-238 released to environs	6.50×10^{-5} gram REDC
	6.50×10^{-6} gram FDPF, FMEF

Source: Calculated results.

I.1.4.1.2 Beyond-Design-Basis Accident

The postulated beyond-design-basis processing facility accident selected for use in this NI PEIS is a catastrophic earthquake. Such an event is less likely than the design-basis processing facility accidents, although its consequences could be severe. Its frequency is assumed to be 1×10^{-5} per year.

CASE 1—STORAGE FACILITY

The earthquake is postulated to collapse the stack, severely damaging the HEPA filter system located nearby. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked or broken. The ventilation systems exhausting from the hot cells are expected to fail. Neptunium-237 is stored in double steel cans, with both the inner and outer cans sealed. The double cans are stacked in an array of robust, seismically supported steel storage tubes inside the hot cell. The analysis postulated the storage tube array would maintain geometry and not be damaged by equipment dislodged within the hot cell during the event. It was postulated that none of the storage cans in the storage tubes would be damaged. The storage cans would not be stressed to a level that would breach the double containment of the can design. No neptunium was postulated to be released from the storage cans during the event.

At INEEL, neptunium-237 may be stored in the CPP–651 vault, which is within 91 meters (100 yards) of FDPF. The CPP–651 vault has 100 in-ground concrete storage silo positions sealed with 5-centimeter (2-inch) stainless steel shielding plugs. The neptunium-237 storage cans would be placed in a rack inside the silo. While the postulated beyond-design-basis earthquake could cause portions of the facility to collapse, none of the storage cans in the in-ground storage silos would be breached. The storage cans would not be stressed to a level that would breach the double containment of the can design.

CASE 2—PROCESSING FACILITY

The earthquake is postulated to collapse the stack, severely damaging the HEPA filter system located nearby. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked or broken. The ventilation systems exhausting from the hot cells are expected to fail. Radioactive materials in the hot cells will be released as a result of cracks in cell walls and shielded windows, but the rate of leakage is expected to be low, since the hot cells are not pressurized and there is no forced ventilation. The leak path factor (i.e., the mass fraction of airborne particulates in an enclosure that is released to the environment) under these conditions has been conservatively estimated to be 0.1 (Green 1997).

The plutonium-238 inventory in the facility will be in several different chemical and physical forms. Since processing is carried on in batches that overlap one another (Wham et al. 1998), the total quantity of plutonium-238 considered available for release from the facility is the sum of the amounts in the dissolver tank, in the ion exchange column during purification, and in powder form and not yet been placed into a sealed canister. Any plutonium-238 in irradiated targets awaiting processing is unlikely to be mechanically damaged by the earthquake because of their rather small size and thus resistance to mechanical breakage. Even if some targets were broken, the plutonium-238 is intimately mixed with the neptunium-237 oxide and an aluminum matrix, rendering it essentially immobile. The earthquake is postulated to result in a massive spill and/or failure of the dissolver tank, an explosion in an ion exchange column, and a spill of any plutonium-238 powder not in a sealed container.

Table I–45 shows the release fractions and the ground-level release source terms for this accident.

Table I–45 Beyond-Design-Basis Earthquake Accident Source Terms

Analysis Parameters	Plutonium-238 Form and Location			
	Solution—Dissolver Tank	Solution—Ion Exchange Column	Powder—Hot Cell Cubicle	Total
Material at risk	1,000 grams	495 grams	186 grams	1,681 grams
Released into hot cell	1,000 grams	495 grams	186 grams	1,681 grams
Airborne release fraction times respirable fraction	0.00013	0.07	0.0033	–
Leak path factor	0.1	0.1	0.1	–
Released to environs	0.013 gram	3.47 grams	0.061gram	3.54 grams

Source: Calculated results.

I.1.4.2 Medical, Industrial, and Research and Development Isotope Processing

The accident analyses included identification of a set of potential accidents that could occur during medical, industrial, and research and development isotope processing at the Radiochemical Processing Laboratory (RPL) (Building 325), FMEF, and the support facility. The analyses are based on scenarios evaluated in the *Building 325 Safety Analysis Report* (the safety analysis report for RPL) for similar types of processes (Battelle 2000). Since process details at the FMEF and the support facility are essentially the same as those at RPL, the same spectrum of accidents was evaluated for all the processing facilities.

The set of accidents evaluated for the Safety Analysis Report was selected using a standard Preliminary Hazards Assessment to identify the potential hazardous conditions in facility operations and to determine the significance of potential accidents. The types of events considered in the Safety Analysis Report included operator errors and handling accidents, fires and explosions, natural phenomena such as seismic events, criticality, and external events such as loss of support services.

For this analysis one bounding event was identified in each of the frequency categories evaluated in the Safety Analysis Report (anticipated, unlikely, or extremely unlikely) in order to identify the events that result in both maximum consequence and maximum risk to onsite and offsite individuals. All types of events that could apply to medical isotope processing were evaluated to determine which scenarios could result in the maximum radionuclide release fraction for each frequency category. The analysis is intended to provide a conservative estimate for the potential consequences of the proposed activities.

Potential accidental releases of radioactive materials during medical, industrial, and research and development isotope processing were estimated using projected radionuclide inventories for the target systems most likely to be considered for production of medical, industrial, and research and development isotopes. **Table I–46** presents the radioactive inventories for the most likely target products. The irradiated targets have a much greater radioactive inventory than the unirradiated targets. Only the radium-226 target is initially radioactive, and its products, actinium and thorium, have greater health consequences than the initial radium-226. Therefore, the accident consequences were analyzed for the irradiated target products.

Processing was assumed to occur 1 day after removal of the targets from the reactor, and only the isotopes associated with a single product target are assumed to be at risk for release in any given event. Release fractions for the radionuclides associated with each product were calculated using the same assumptions as those used for similar types of materials in the Safety Analysis Report scenarios.

Table I–46 Medical, Industrial, and Research and Development Target Product Inventories

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Gold-198	Gold-198	132
R3	Gold-198m	0.00
	Gold-199	150
	Gold-200	0.00816
	Gold-200m	0.0434
	Mercury-203	1.79×10^{-8}
	Platinum-197	9.83×10^{-6}
Cadmium-109	Cadmium-109	654
LTIV	Sodium-108	3.53×10^{-5}
	Sodium-108m	3.92×10^{-4}
	Sodium-110	0.00878
	Sodium-110m	0.627
	Sodium-111	0.119
	Sodium-111m	4.11×10^{-12}
	Palladium-109	2.73×10^{-4}
Copper-64	Copper-64	1,300
R3	Copper-66	2.50×10^{-14}
	Copper-67	4.36×10^{-7}
	Nickel-65	6.42×10^{-9}
	Nickel-66	8.80×10^{-4}
	Zinc-65	138
Copper-67	Copper-67	6.26
R3	Copper-66	0.00
	Copper-68	1.98×10^{-13}
	Copper-69	8.88×10^{-31}
	Zinc-65	0.00
	Zinc-69	0.00268
	Zinc-69m	0.00268
Gadolinium-153	Gadolinium-153	1,100
LTIV	Europium-152	4,660
	Europium-152m	6.41×10^{4}
	Europium-154	1.55×10^{4}
	Europium-154m	2.20×10^{-4}
	Europium-155	3,540
	Europium-156	3.39×10^{5}
	Samarium-153	3.16×10^{4}
Holmium-166	Holmium-166	58.9
R3	Dysprosium-166	2.07×10^{-6}
	Dysprosium-167	8.80×10^{-20}
	Erbium-167m	1.91×10^{-4}
	Erbium-169	2.13×10^{-6}
	Holmium-166m	9.92×10^{-5}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Holmium-167	1.91×10^{-4}
	Holmium-168	3.12×10^{-19}
	Holmium-169	8.29×10^{-36}
	Holmium-170	0.00
	Holmium-170m	0.00
Samarium-145	Samarium-145	11.8
LTIV	Neodymium-147	1.54×10^{-5}
	Neodymium-149	3.72×10^{-20}
	Neodymium-151	3.36×10^{-25}
	Promethium-145	1.21
	Promethium-146	3.50×10^{-4}
	Promethium-147	0.00162
	Promethium-148	0.00365
	Promethium-148m	8.70×10^{-4}
	Promethium-149	0.00218
	Promethium-150	2.56×10^{-7}
	Promethium-151	1.89×10^{-9}
	Samarium-151	2.38×10^{-7}
Samarium-153	Samarium-153	70.7
R3	Europium-152	0.00
	Europium-152m	0.00
	Europium-154	0.00348
	Europium-154m	8.75×10^{-10}
	Europium-155	2.32×10^{-4}
	Europium-156	0.00446
	Gadolinium-153	0.00
	Samarium-151	0.00
	Samarium-155	8.63×10^{-14}
	Samarium-156	1.11×10^{-9}
Tin-117m	Tin-117m	48.5
R3	Antimony-122	0.00118
	Antimony-122m	4.72×10^{-8}
	Tin-119m	3.92×10^{-8}
	Tin-121	0.00
	Tin-121m	4.76×10^{-11}
Strontium-85	Strontium-85	2,160
LTIV	Krypton-83m	0.00
	Krypton-85	9.51×10^{-4}
	Krypton-85m	1.50×10^{-4}
	Rubidium-83	0.00
	Rubidium-84	8.48
	Rubidium-86	4.18
	Strontium-83	0.00
	Strontium-85m	0.00101
	Strontium-89	9.40×10^{-9}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Strontium-90	6.09×10^{-15}
Strontium-89	Strontium-89	156
LTIV	Krypton-85	0.00
	Krypton-85m	0.00
	Krypton-87	0.00
	Rubidium-84	0.00
	Rubidium-86	0.00
	Rubidium-86m	0.00
	Rubidium-88	2.09×10^{-12}
	Strontium-85	0.00
	Strontium-87m	0.00
	Strontium-90	1.69×10^{-4}
	Yttrium-88	2.71×10^{-6}
	Yttrium-90	0.0218
	Yttrium-90m	2.08×10^{-7}
	Zirconium-89	8.15×10^{-9}
Iodine-125	Iodine-125	2,530
Gas Line	Iodine-124	0.00
	Iodine-126	0.00
	Xenon-125	0.00
Iodine-131	Iodine-131	307
R3	Iodine-132	0.00867
	Iodine-132m	3.52×10^{-7}
	Tellurium-131	3.46
	Tellurium-131m	15.5
	Tellurium-132	0.00830
	Xenon-131m	2.02
Iridium-192	Iridium-192	3,570
LTIV	Iridium-192m	8.36×10^{-8}
	Iridium-193m	27.6
	Iridium-194	0.0317
	Iridium-194m	0.00991
	Iridium-195	1.10×10^{-6}
	Iridium-195m	3.55×10^{-6}
	Platinum-193	0.0886
	Platinum-193m	13.2
	Platinum-195m	3.82×10^{-4}
Lutecium-177	Lutecium-177	0.519
R3	Hafnium-177m	0.00111
	Hafnium-178m	9.88×10^{-13}
	Hafnium-179m	3.10×10^{-9}
	Lutecium-176m	0.00
	Lutecium-177m	0.00143
	Lutecium-178	9.87×10^{-13}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Lutecium-179	3.09×10^{-9}
Molybdenum-99	Molybdenum-99	1,680
R3	Molybdenum-101	2.08×10^{-5}
	Molybdenum-102	3.79×10^{-18}
	Molybdenum-103	3.71×10^{-34}
	Ruthenium-103	2.09×10^{-6}
	Technetium-99m	1,830
	Technetium-100	8.94×10^{-10}
	Technetium-101	2.08×10^{-5}
	Technetium-102	3.79×10^{-18}
	Technetium-103	2.84×10^{-34}
Osmium-194	Osmium-194	2.20
LTIV	Iridium-192	0.00
	Iridium-192m	0.00
	Iridium-193m	0.00
	Iridium-194	2.19
	Iridium-194m	0.00
	Osmium-190m	0.00
	Osmium-191	0.00
	Osmium-191m	0.00
	Osmium-192m	0.00
	Osmium-193	9.02×10^{4}
	Rhenium-189	0.00
	Rhenium-190	0.00
	Rhenium-190m	0.00
	Rhenium-191	0.00
Tungsten-188	Tungsten-188	5,810
LTIV	Hafnium-181	0.00
	Hafnium-182m	0.00
	Osmium-189m	140
	Rhenium-186	0.00
	Rhenium-188	4.45×10^{4}
	Rhenium-188m	1.86×10^{-7}
	Rhenium-189	46.4
	Tantalum-182	0.00
	Tantalum-182m	0.00
	Tantalum-183	0.00
	Tungsten-181	0.00
	Tungsten-185	0.00
	Tungsten-185m	0.00
	Tungsten-187	7.24×10^{5}
	Tungsten-189	3.69×10^{-9}
Xenon-127	Xenon-127	7.26
LTIV		

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Yttrium-91	Yttrium-91	17.8
LTIV	Yttrium-90	0.00
	Zirconium-89	0.00
	Zirconium-95	2.88×10^{-19}
Phosphorus-32	Phosphorus-32	39.1
R3	Phosphorus-33	0.0518
	Phosphorus-34	1.63×10^{-14}
	Phosphorus-35	2.92×10^{-32}
	Phosphorus-36	0.00
	Sulfur-35	1.99×10^{-8}
Phosphorus-33	Phosphorus-33	76.2
LTIV	Argon-37	1.88×10^{-25}
	Phosphorus-32	0.00
	Sulfur-35	7.29×10^{-6}
Palladium-103	Palladium-103	1,340
R3	Silver-107m	1.58×10^{-26}
	Palladium-107m	2.00×10^{-20}
	Rhodium-103m	1,350
	Rhodium-104	2.74×10^{-9}
	Rhodium-104m	1.89×10^{-10}
	Rhodium-105	9.07×10^{-6}
	Rhodium-105m	2.13×10^{-26}
	Rhodium-106	2.45×10^{-18}
	Rhodium-106m	3.86×10^{-11}
	Rhodium-107	5.24×10^{-26}
Platinum-195m	Platinum-195m	168
R3		
Rhenium-186	Rhenium-186	4,350
R3	Osmium-189m	1.48×10^{-4}
	Osmium-190m	1.23×10^{-11}
	Rhenium-188	0.0550
	Rhenium-188m	2.60×10^{-13}
	Rhenium-189	1.16×10^{-5}
	Rhenium-190	5.39×10^{-12}
	Rhenium-190m	1.15×10^{-11}
	Tungsten-187	6.41
	Tungsten-188	0.0113
	Tungsten-189	9.74×10^{-15}
Scandium-47	Scandium-47	29.6
R3	Calcium-45	0.00
	Calcium-47	1.82×10^{-5}
	Scandium-46	0.00
	Scandium-48	0.0202

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Selenium-75	Selenium-75	17.9
LTIV	Arsenic-76	0.114
	Arsenic-77	3.78×10^{-5}
	Arsenic-78	3.94×10^{-13}
	Bromine-80	0.00
	Bromine-80m	2.92×10^{-36}
	Selenium-77m	1.82×10^{-13}
	Selenium-79m	9.30×10^{-18}
Actinium-227	Actinium-227	34.0
LTIV	Actinium-228	56.1
	Actinium-229	6.04×10^{-9}
	Radium-226	14.3
	Radium-227	4.23×10^{-7}
	Radium-228	0.00101
	Radium-229	5.00×10^{-14}
	Thorium-227	24.8
	Thorium-228	42.1
	Thorium-229	8.63×10^{-4}
	Actinium-225	3.72×10^{-4}
	Astatine-217	3.72×10^{-4}
	Bismuth-210	0.109
	Bismuth-211	19.6
	Bismuth-212	24.6
	Bismuth-213	3.71×10^{-4}
	Bismuth-214	14.3
	Francium-221	3.72×10^{-4}
	Francium-223	1.40×10^{-5}
	Lead-209	3.69×10^{-4}
	Lead-210	0.118
	Lead-211	19.6
	Lead-212	38.4
	Lead-214	14.3
	Polonium-210	0.106
	Polonium-211	0.0535
	Polonium-212	24.6
	Polonium-213	3.63×10^{-4}
	Polonium-214	14.3
	Polonium-215	19.6
	Polonium-216	38.8
	Polonium-218	14.3
	Radium-223	19.6
	Radium-224	38.8
	Radium-225	5.46×10^{-4}
	Radon-217	4.46×10^{-8}
	Radon-219	19.6
	Radon-220	38.8
	Radon-222	14.3

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Thallium-207	19.6
	Thallium-208	8.83
	Thallium-209	8.16×10^{-6}

a. Assumes a 1-day cooling time after irradiation.
Key: LTIV, Long-Term Irradiation Vehicle; R^3, Rapid Radioisotope Retrieval system.
Source: BWHC 1999.

I.1.4.2.1 Localized Solvent Fire

The safety analysis report for RPL identified a number of accident scenarios with an anticipated frequency greater than 0.01 per year. The types of accidents that fell into this category included the following:

1. Localized solvent fire
2. Localized solid fire
3. Spill in a hot cell
4. Spill in a laboratory

Of these events, the scenario with the highest radionuclide release was the solvent fire. A localized fire of sufficient severity to produce radionuclide releases was estimated to occur no more than once in 10 years. The upper-bound frequency of such an event was supported by the fire loss history at Hanford over a 45-year period. During that time, the site experienced 10 fires that resulted in significant property loss. Of those fires, 6 potentially involved radioactive materials, and 2 of the 6 occurred in laboratory facilities. No fires of that magnitude have occurred in RPL since it was occupied in 1953 and would not be expected to occur routinely in that facility because of the facility design, administrative controls on conduct of operations, and the fire protection program. Since only 2 events potentially involving radioactive materials occurred in laboratory facilities over a 45-year period, a frequency of 4.44×10^{-2} per year was assumed for this accident.

The heating, ventilating, and air conditioning system was assumed to be operating during and after the fire. Combustibles (e.g., solvent-soaked rags) were assumed to be present in sufficient quantity to support combustion. The source terms used for this accident scenario were based on radioactive materials representative of anticipated medical-isotope-processing activities in the hot cells and other laboratories in the facility. Manual fire suppression was assumed not to occur or to be ineffective.

The final HEPA filters were assumed to be unaffected by the fire because they are located in a facility separate from RPL. This assumption was based on the observations that the primary filters would stop most smoke particles and that air dilution would cool the hot gases leaving the laboratory or hot cell so the final HEPA filter bank would not be subjected to extreme temperatures. Therefore, the final stage of HEPA filters was assumed to remain intact. For conservatism, particle deposition along the release path was assumed not to occur. The radon holdup system was assumed to be ineffective, and it was also assumed that no deposition or filtration of noble gases would occur.

It was assumed that charcoal filters will be included in the emergency ventilation system. The activated-charcoal filters will comply with current industry standards. Filter efficiency was conservatively assumed to be 99 percent, consistent with NRC Regulatory Guide 1.52 (NRC 1978). For conservatism, iodine deposition prior to filtration was assumed not to occur.

The radionuclide releases for this event, as estimated in the safety analysis report, were assumed to be the same as those for a solvent fire involving radioactive solutions. Although many of the processes for preparing medical and industrial isotopes would involve only dissolution in aqueous acid solution, some of the chemical

separations could require solvent extraction or ion exchange apparatus. Therefore, the solvent fire was assumed to be a bounding case for this scenario. Separate release fractions were calculated for nonvolatile materials, volatile materials (iodine, sulfur), and noble gases.

The release fractions were calculated by the following generic formula: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. Calculations for the three releases fractions are:

Nonvolatiles: $0.01 \times 1.0 \times 1.0 \times 0.0005 = 5\times10^{-6}$

Volatiles: $1.0 \times 1.0 \times 1.0 \times 0.01 = 0.01$

Gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

To determine which irradiated target would result in the maximum consequences, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions. The resulting inventories were then multiplied by dose conversion factors. This final multiplication resulted in a dose for each isotope. The isotope doses within each target were totaled for a target dose. The target doses were compared to determine the target with the maximum dose consequences. The iodine-125 gas line product target resulted in the highest target dose. Therefore, the iodine-125 product target was used to determine the bounding consequences for the localized fire event.

The resulting source term for the localized fire accident is an elevated release of 25.3 curies of iodine-125.

I.1.4.2.2 Unlikely Seismic Event

Events in the unlikely frequency category (between 1×10^{-4} and 1×10^{-2} per year) in the RPL safety analysis report included:

1. Liquid waste cask failure and spill
2. Unlikely seismic event

Of these scenarios, the unlikely seismic event resulted in a higher radionuclide release fraction. This event was assigned to the unlikely frequency category due to the return period of the initiating earthquake. Earthquake hazard curves have been developed for the 300 Area that define ground acceleration at RPL for a given frequency. The seismic event analyzed in this section has a peak horizontal ground acceleration of 0.139 g for a frequency range of 1×10^{-4} to 1×10^{-2} per year. For earthquakes in the unlikely category, a single potential process upset was assumed, but it was estimated that multiple major upsets would not occur. The facility's superstructure was assumed to remain intact, but the heating, ventilating, and air conditioning system was assumed to fail because it has not been seismically qualified. For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

Spilling of the powdered contents of one in-process medical, industrial, or research and development isotope target was conservatively assumed to occur (i.e., the probability of the spill given that the seismic event occurs was assumed to be 1.0). The release from the spill and holdup release were reduced by 50 percent to account for deposition of the powder within the facility. This 50 percent building removal factor could be applied to this scenario because of essentially static conditions that result from failure of the ventilation system. Because this event does not involve a heat source to mobilize volatile materials, the release fraction was assumed to be the same for all materials except noble gases. The radon holdup system was assumed to be ineffective for this scenario.

Releases from this event were estimated as follows: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. The specific release fractions are:

Nongases: $0.002 \times 0.3 \times 0.5 \times 1.0 = 3\times10^{-4}$

Gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

As in the anticipated category source term analysis, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions and dose conversion factors. A comparison of the target doses indicated that the actinium-227 product target results in maximum dose consequences for this accident. In addition to the actinium-227 product, the irradiated target contains 9 impurity isotopes and 32 decay products. The dose screening, however, determined that at least 99.9 percent of the total dose is attributable to six isotopes (actinium-227, radium-223, radium-224, radium-226, thorium-227, thorium-228).

The ground-level release source term for the unlikely seismic event is as follows (releases in curies):

Isotope	Release
Actinium-227	0.0102
Radium-223	0.00588
Radium-224	0.0116
Radium-226	0.00429
Thorium-227	0.00744
Thorium-228	0.0126

I.1.4.2.3 Loss of Electric Power and Explosion

The safety analysis report for RPL identified the following events in the extremely unlikely category (between 1×10^{-6} and 1×10^{-4} per year):

1. Loss of electric power and explosion
2. Large uncontrolled fire
3. Extremely unlikely seismic event

Of the extremely unlikely events identified in the safety analysis report for RPL, the highest radionuclide release was associated with the loss of electrical services followed by an explosion. This scenario assumes loss of power to RPL, which inactivates the ventilation system. On failure of the ventilation systems, airflow through the hot cells, gloveboxes, hoods, and tanks would also cease. Without ventilation, the potential exists for a buildup of flammable or combustible vapors in those areas with volatile chemicals. A deflagration in a glovebox from the buildup of a flammable solvent or volatile chemical is assumed to occur, potentially breaching the primary confinement barriers. The walls and ceilings of the glovebox or fume hood would mitigate the impact of an explosion. Most of the airborne material within a glovebox or hood would be carried out through the exhaust system, even if the explosion were to cause material to be released from the glovebox to an adjoining area. Judging from actual glovebox explosions, the front panel of a glovebox could fail. In most cases, these events have not resulted in offsite releases because the explosions did not cause malfunctions of ventilation systems or the failure of other barriers, including room walls and ceilings. However, for the extremely unlikely scenario in this analysis, the explosion is assumed to be sufficiently forceful to breach the building or ventilation system barriers, rendering the HEPA filters and radon holdup system ineffective.

Releases from this event were estimated as follows: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. The specific release fractions are:

Nonvolatiles: $0.05 \times 0.4 \times 0.5 \times 1.0 = 0.01$

Volatiles and gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

As in the previous source term analyses, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions and dose conversion factors. A comparison of the target doses indicated that the actinium-227 product target results in maximum dose consequences for this accident.

The ground-level release source term for the extremely unlikely loss of electrical services followed by an explosion is:

Isotope	Release
Actinium-227	0.340
Radium-223	0.196
Radium-224	0.388
Radium-226	0.143
Thorium-227	0.248
Thorium-228	0.421

I.1.5 Methodology for Estimating Processing Facility Accident Radiological Impacts

The exposure, uptake, and usage parameters used in the GENII model for assessing processing facility accident impacts are provided in **Tables I–47** through **I–49**. The GENII computer code was used to estimate the radiological consequences of the postulated accidents at the processing facilities. A discussion of the GENII computer code is presented in Appendix H. Doses to a noninvolved worker, the maximally exposed individual, and the population within 80 kilometers (50 miles) of each plant were calculated. To determine the consequences for the maximally exposed individual, doses were calculated at the site boundary and at the nearest highway within the site boundary. The population and boundary maximally exposed individual doses included doses via ingestion. The nearest highway maximally exposed individual is assumed to be exposed for a period of 2 hours. The consequences (doses) were then multiplied by the frequencies of the accidents to determine the risk.

Table I–47 GENII Exposure Parameters to Plumes and Soil Contamination (Accidents)

Maximum Individual				General Population			
External Exposure		Inhalation of Plume		External Exposure		Inhalation of Plume	
Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cubic centimeters per second)	Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cubic centimeters per second)
0.00	6,136	100% of release time	330	0.00	4,383	100% of release time	330

Source: Napier et al. 1988; NRC 1977.

Table I–48 GENII Usage Parameters for Consumption of Terrestrial Food (Accidents)

Food Type	Maximum Individual				General Population			
	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)
Leafy vegetables	90.0	1.5	1.0	30.0	90.0	1.5	14.0	15.0
Root vegetables	90.0	4.0	5.0	220.0	90.0	4.0	14.0	140.0
Fruit	90.0	2.0	5.0	330.0	90.0	2.0	14.0	64.0
Grains/cereals	90.0	0.8	180.0	80.0	90.0	0.8	180.0	72.0

Key: kg/m^2, kilograms per square meter; kg/yr, kilograms per year.
Source: Napier et al. 1988.

Table I–49 GENII Usage Parameters for Consumption of Animal Products (Accidents)

Food Type	Consumption Rate (kg/yr)	Holdup Time (days)	Stored Feed				Fresh Forage			
			Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)	Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)
Maximum individual										
Beef	80.0	15.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	18.0	1.0	1.00	90.0	0.80	180.0	--	--	--	--
Milk	270.0	1.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	30.0	1.0	1.00	90.0	0.80	180.0	--	--	--	--
General population										
Beef	70.0	34.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	8.5	34.0	1.0	90.0	0.80	180.0	--	--	--	--
Milk	230.0	3.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	20.0	18.0	1.0	90.0	0.80	180.0	--	--	--	--

Key: kg/yr, kilograms per year; kgs/m^2, kilograms per square meter.
Source: Napier et al. 1988.

It is possible that an individual member of the public could be closer to a facility than either the site boundary or the nearest onsite highway. However, such individuals would be present only occasionally and for brief periods (a few hours or more). Hence, the annual probability that an individual would be close is relatively low, and the associated risk to that individual would be bounded by the site boundary or onsite highway maximally exposed individual risk.

For all of the processing facilities, accidents were evaluated using 50 percent meteorology. The meteorological data used were in the form of joint frequency data files at each site location. The joint frequency data files were based on measurements taken over a period of several years at different locations and heights. Population distributions were based on the 1990 Census of Population and Housing (DOC 1992). Projections were determined for the year 2020 (approximate midlife of operations) for areas within 80 kilometers (50 miles) of each facility.

Consequences to involved workers are addressed in Section I.1.7.

In addition to the GENII computer code, the MACCS2 computer code was used for the localized solvent fire accident analysis. GENII does not permit interdiction or the disposal of food and therefore is inappropriate for the ingestion pathway analysis for the solvent fire accident. The iodine release postulated during the fire accident is sufficient to prompt interdiction. Therefore, GENII was used for the inhalation and external exposure pathways, and MACCS2 was used for the ingestion pathway. The doses via the two pathways were then summed for the total dose.

I.1.6 Processing Facility Accident Consequences and Risks

The processing facility accident source term data presented in Sections I.1.4.1 and I.1.4.2 were evaluated using the GENII accident analysis computer code. The MACCS2 computer code was used in addition to GENII for the medical, industrial, and research and development isotope localized solvent fire accident. **Tables I–50** through **I–54** summarize the consequences and risks of the accidents, with mean meteorological conditions, for the maximally exposed individual, the offsite population within an 80-kilometer (50-mile) radius of the facility, and a noninvolved worker 640 meters (2,100 feet) from the release point.

Table I–50 presents REDC accident consequences and risks for plutonium-238 fabrication and processing.

Table I–51 presents FDPF accident consequences and risks for plutonium-238 fabrication and processing.

Table I–52 presents FMEF accident consequences and risks for plutonium-238 fabrication and processing and for simultaneous plutonium-238, medical, industrial, and research and development isotope fabrication and processing.

Table 1–53 presents RPL accident consequences and risks for medical, industrial, and research and development isotope fabrication and processing.

Table I–54 presents generic support facility accident consequences and risks for medical, industrial, and research and development isotope fabrication and processing.

Table I–50 REDC Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Ion exchange explosion during neptunium-237 target fabrication (0.01)	6.13×10^{-9}	3.06×10^{-12}	3.06×10^{-14}	8.58×10^{-5}	4.29×10^{-8}	4.29×10^{-10}	5.60×10^{-10}	2.24×10^{-13}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	1.76×10^{-7}	8.79×10^{-11}	8.79×10^{-13}	0.00196	9.82×10^{-7}	9.82×10^{-9}	1.69×10^{-8}	6.74×10^{-12}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	4.68×10^{-4}	2.34×10^{-7}	2.34×10^{-9}	5.23	0.00261	2.61×10^{-5}	4.49×10^{-5}	1.79×10^{-8}	1.79×10^{-10}
Plutonium-238 processing facility beyond design-basis earthquake (1×10^{-5})	163	0.163	1.63×10^{-6}	8.91×10^{5}	445	0.00445	1,310	1.00[e]	1.00×10^{-5}[f]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: REDC, Radiochemical Engineering Development Center.
Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–51 FDPF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Ion exchange explosion during neptunium-237 target fabrication (0.01)	2.01×10^{-9}	1.01×10^{-12}	1.01×10^{-14}	2.49×10^{-5}	1.24×10^{-8}	1.24×10^{-10}	7.26×10^{-9}	2.91×10^{-12}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	6.11×10^{-8}	3.05×10^{-11}	3.05×10^{-13}	5.65×10^{-4}	2.82×10^{-7}	2.82×10^{-9}	2.17×10^{-7}	8.69×10^{-11}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	1.63×10^{-5}	8.13×10^{-9}	8.13×10^{-11}	0.150	7.51×10^{-5}	7.51×10^{-7}	5.79×10^{-5}	2.31×10^{-8}	2.31×10^{-10}
Plutonium-238 processing facility beyond design-basis earthquake (1×10^{-5})	42.5	0.0425	4.25×10^{-7}	1.64×10^{5}	82.0	8.20×10^{-4}	1,200	1.00[e]	1.00×10^{-5}[f]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: FDPF, Fluorinel Dissolution Process Facility.
Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–52 FMEF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Plutonium-238 processing									
Ion Exchange explosion during neptunium-237 target fabrication (0.01)	2.02×10^{-9}	1.01×10^{-12}	1.01×10^{-14}	7.26×10^{-5}	3.63×10^{-8}	3.63×10^{-10}	6.65×10^{-10}	2.66×10^{-13}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	4.64×10^{-8}	2.32×10^{-11}	2.32×10^{-13}	0.00169	8.47×10^{-7}	8.47×10^{-9}	1.95×10^{-8}	7.81×10^{-12}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	1.24×10^{-5}	6.18×10^{-9}	6.18×10^{-11}	0.451	2.25×10^{-4}	2.25×10^{-6}	5.20×10^{-6}	2.08×10^{-9}	2.08×10^{-11}
Plutonium-238 processing only									
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	16.5	0.00823	8.23×10^{-8}	6.41×10^{5}	321	0.00321	921	1.00[e]	1.00×10^{-5}[(f)]
Medical and industrial isotope processing									
Medical and industrial isotopes localized solvent fire (0.044)	0.00276	1.38×10^{-6}	6.13×10^{-8}	56.2	0.0281	0.00125	9.51×10^{-5}	3.80×10^{-8}	1.69×10^{-9}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	1.00	5.00×10^{-4}	5.00×10^{-8}	2.95×10^{4}	14.8	0.00148	24.0	0.0192	1.92×10^{-6}
Medical and industrial isotope and plutonium-238 processing									
Processing facility beyond-design-basis earthquake (1×10^{-5})	16.5	0.00825	8.25×10^{-8}	6.42×10^{5}	321	0.00321	922	1.00[e]	1.00×10^{-5}[(f)]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those e×posed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: FMEF, Fuels and Materials Examination Facility.

Source: Model results, using the GENII computer code (Napier et al. 1998) and the MACCS2 computer code (Chanin and Young 1997).

Table I–53 RPL Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Medical and industrial isotopes localized solvent fire (0.044)	0.0135	6.74×10^{-6}	2.99×10^{-7}	77.8	0.0389	0.00173	0.00470	1.88×10^{-6}	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	1.52	7.60×10^{-4}	7.60×10^{-6}	1,350	0.675	0.00675	1.50	6.00×10^{-4}	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	50.0	0.050	5.00×10^{-6}	4.60×10^{4}	23.0	0.00230	49.0	0.0392	3.92×10^{-6}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: RPL, Radiochemical Processing Laboratory.

Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–54 Generic Support Facility Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Medical and industrial localized solvent fire (0.044)	0.0194	9.72×10^{-6}	4.32×10^{-7}	31.1	0.0156	6.91×10^{-4}	0.00530	2.12×10^{-6}	9.41×10^{-8}
Medical and industrial unlikely seismic event (0.01)	0.0750	3.75×10^{-5}	3.75×10^{-7}	136	0.0680	6.80×10^{-4}	0.510	2.04×10^{-4}	2.04×10^{-6}
Medical and industrial glovebox explosion (1.00×10^{-4})	2.50	0.00125	1.25×10^{-7}	4,600	2.30	2.30×10^{-4}	17.0	0.00680	6.80×10^{-7}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results, using the GENII (Napier et al. 1988) and MACCS2 (Chanin and Young 1997) computer codes.

I.1.7 Involved Worker Accident Consequences and Risks

The estimated number of involved workers at each of the proposed irradiation and processing facilities is shown in **Table I–55**.

Table I–55 Estimated Number of Involved Workers at Each Facility

Facility	Type	Involved Workers
FFTF	Irradiation	242
ATR	Irradiation	100
HFIR	Irradiation	100
CLWR	Irradiation	300
New research reactor	Irradiation	120
High-energy accelerator	Irradiation	225
Low-energy accelerator	Irradiation	13
REDC	Processing	116
FDPF	Processing	75
FMEF	Processing	105
RPL/306–E	Processing	30
New support facility	Processing	100

I.1.7.1 Irradiation Facility Consequences and Risks

I.1.7.1.1 Design-Basis Accident

Each of the proposed irradiation facilities would have an approved onsite emergency plan. The likelihood of a design-basis accident is estimated to be once in 10,000 years. Since an accident could occur at any given time, the number of workers on site at the time of an accident would be unlikely to exceed one-third the total number of involved workers shown above (assuming a three-shift operation). The workers at the facility at the time of a design-basis accident can be grouped into three major categories, as follows:

1. Those workers not having duties associated with accident management or recovery. These would be promptly notified and evacuated from the site. Individuals in this group would be expected to receive low doses significantly below the EPA Protective Action Guides (PAGs) (EPA 1992) of 1-5 rem. Most involved workers would be in this group.

2. Those workers located in shielded areas such as the control room or other designated plant emergency operation areas having duties associated with accident management and recovery. These workers, because of the radiation protection afforded by their locations, would be unlikely to receive doses in excess of the EPA PAGs. For the irradiation facilities, this group is estimated to range from about 6 to 20 individuals.

3. Those few workers in areas of the plant who may be directly affected or impacted by the accident, (e.g., performing maintenance in the immediate area where an accident initiating event occurs). This very small group of involved workers could receive significant doses in excess of the EPA PAGs. With appropriate radiation instrumentation, alarms and administrative controls, it is unlikely that individuals in this group would receive doses high enough to result in acute radiation effects (doses greater than about 100 rem).

I.1.7.1.2 Beyond-Design-Basis Accident

The likelihood of a beyond-design-basis accident is estimated to be once in 100,000 years. A beyond-design-basis accident may begin as a design-basis accident, but would involve additional equipment failures that lead to more serious reactor or facility damage than in a design basis event. For this reason, the previous discussion on design-basis events is largely applicable. Most involved workers would likely be evacuated prior to receiving any significant dose. A small group of workers, including operators and personnel directly involved in accident management and recovery could receive significant doses, however. One or two individuals could conceivably receive high doses if emergency actions were to be taken (e.g., entering a high radiation area for a short time to actuate a valve or pump).

I.1.7.2 Processing Facility Consequences and Risks

I.1.7.2.1 Design-Basis Accident

GLOVEBOX EXPLOSION DURING NEPTUNIUM-237 TARGET FABRICATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. Assuming this accident occurs, the involved worker at the affected glovebox may be seriously injured as a result of the explosion as well as likely to be contaminated with the explosion debris. This worker could receive a significant radiation dose. The extent of the contamination and the radiation doses are likely to be highly localized, however. Neighboring workers in nearby gloveboxes will be exposed to significantly lower doses and effects from the explosion, while workers in other locations in the processing facility will be only minimally affected.

TARGET DISSOLVER TANK FAILURE DURING PLUTONIUM-238 SEPARATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. This accident is postulated to occur in a shielded hot cell, whose integrity is not challenged by this accident. Consequently, those workers outside the hot cell carrying out this operation will not be affected. Some plutonium-238 and neptunium-237 will be released from an elevated stack after passing through two stages of HEPA filters, which removes all but a very small fraction of the spilled tank contents. Workers at the processing facility will be exposed to very low concentrations of plutonium-238 and neptunium-237 as a result of this release. It is estimated that since an elevated release results in very low concentrations at ground level close to the release point, that worker doses will be generally similar to those received by the maximally exposed offsite individual for this accident.

ION EXCHANGE EXPLOSION DURING PLUTONIUM-238 SEPARATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. As for the postulated dissolver tank failure discussed above, this accident is postulated to occur in a shielded hot cell, whose integrity is unlikely to be challenged by this accident. Consequently, those workers outside the hot cell carrying out this operation will not be affected. Some plutonium-238 will be released from an elevated stack after passing through two stages of HEPA filters, which removes all but a very small fraction of the explosion debris. Workers at the processing facility will be exposed to very low concentrations of plutonium-238 as a result of this release. It is estimated that since an elevated release results in very low concentrations at ground level close to the release point, that worker doses will be generally similar to those received by the maximally exposed offsite individual for this accident.

I.1.7.2.2 Beyond-Design-Basis Accident

The beyond-design-basis accident postulated for the processing facilities is a catastrophic earthquake whose likelihood is taken to be once in 100,000 years for this NI PEIS. The earthquake is postulated to collapse the stack, severely damaging the HEPA filters. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked and broken. The ventilation systems exhausting from the hot cells are expected to fail. Radioactive materials in the hot cells will be released as a result of cracks in cell walls and shielded windows, but the rate of leakage is expected to be low, since the hot cells are not pressurized and there is no forced ventilation.

Many of the workers in the processing facility are expected to be injured as a direct result of the earthquake. Those workers who are mobile are expected to leave the facility, and this group is not likely to receive any significant radiation dose. Workers who are trapped in the rubble and debris of the earthquake and unable to leave the immediate vicinity could receive significant additional radiation doses, however.

I.1.7.3 Medical and Industrial Isotopes Processing Facility Consequences and Risks

I.1.7.3.1 Localized Solvent Fire

This event postulates a localized solvent fire in a hot cell. Because only two events potentially involving radioactive materials have occurred in laboratory facilities over a 45-year period, a frequency of 4.44×10^{-2} per year was assumed for this accident. The integrity of the hot cell would not be challenged by the localized nature of the fire, and the final HEPA and iodine-removing charcoal filters were assumed to be unaffected by the fire because they are located away from the hot cells. Some radioactive materials would be released after passing through the filters, which would remove all but a small fraction of the materials reaching them. Workers at the processing facility would be unaffected by the fire, but would be exposed to low concentrations of released radioactive isotopes. It is estimated that worker doses would generally be similar to those received by the maximally exposed individual (0.0135 rem for RPL and 0.0194 rem for the generic support facility) for this accident.

I.1.7.3.2 Unlikely Seismic Event

For the purposes of this NI PEIS, this accident frequency is assumed to be 1×10^{-2} per year. The facility's superstructure was assumed to remain intact, but the heating, ventilating, and air-conditioning system was assumed to fail because it has not been seismically qualified. Some injuries may occur because of the seismic event itself. Releases of radioactive materials would be at ground level. It is estimated that worker doses would generally be somewhat greater than those received by the maximally exposed individual (1.52 rem for RPL and 0.075 rem for the generic support facility) for this accident.

I.1.7.3.3 Loss of Electrical Power and Explosion

For the purposes of this NI PEIS, this accident frequency is assumed to be 1×10^{-4} per year. This scenario assumes a loss of electrical power that inactivates the ventilation system. This is postulated to lead to an explosion in a glovebox as a result of a buildup of flammable vapors. Most of the airborne material within a glovebox would be carried out through the exhaust system. For this scenario, the explosion is assumed to be sufficiently forceful to render the HEPA filters and radon holdup system ineffective. Assuming this accident occurs, the involved worker at the affected glovebox may be seriously injured as a result of the explosion and may be contaminated with explosion debris. This worker could receive a significant radiation dose. The extent of the contamination and the radiation dose is likely to be highly localized, however. Neighboring

workers in nearby gloveboxes, if any, would be exposed to significantly lower doses and effects from the explosion. Releases of radioactive materials would be at ground level. It is estimated that worker doses at the facility (except for the worker directly affected by the explosion) would generally be somewhat greater than those received by the maximally exposed individual (50.0 rem for RPL and 2.5 rem for the generic support facility) for this accident.

I.1.8 Risk Summation

To provide a better indication of risks of the postulated accidents, the risks are summed for each facility and also for each option. The summed risks for each alternative and option are presented in **Tables I–56** through **I–76**. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk of accidents, a full-scope (Level 3) probabilistic risk analysis is required for each facility. However, since full-scope probabilistic risk analyses are not available to incorporate into this NI PEIS, summation of the spectrum of accident risks was considered appropriate.

As explained previously, a full spectrum of accidents was considered at the irradiation and fabrication and processing facilities. The accidents evaluated represent bounding cases that are considered to envelop the risk profile.

For each option, the highest risks are presented for the maximally exposed individual and the noninvolved worker. The highest risk to an individual may result from either a single facility or a combination of facilities. A combination of facilities can occur only if the facilities are colocated. In this case, the individual risks are summed. For each option, all facility population risks are summed.

For the currently operating reactors (ATR, HFIR, and CLWR), the incremental risk of target irradiation is determined by subtracting the risk without target irradiation from the risk with target irradiation. For example, in Alternative 2, Option 1, there is an incremental risk to the maximally exposed individual from a large-break loss-of-coolant accident. The incremental risk is determined by subtracting the maximally exposed individual risk from a large-break loss-of-coolant accident without target irradiation (i.e., 0 kilograms per year plutonium-238 production) from the maximally exposed individual risk with target irradiation (i.e., 5 kilograms per year plutonium-238 production). The incremental risk is therefore 3.02×10^{-8} - 2.33×10^{-8} = 6.09×10^{-9} as presented in the table. The incremental risks are used to determine the summed risks. Therefore, summing every risk presented in the tables will not directly produce the summed risk.

Table I–56 Risk Summation for Alternative 1—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{4.51\times10^{-4}}$	$\mathbf{5.35\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–57 Risk Summation for Alternative 1—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 2	$\mathbf{4.51\times10^{-4}}$	$\mathbf{4.07\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–58 Risk Summation for Alternative 1—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
FMEF (full processing)			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotope localized solvent fire	6.13×10^{-8}	1.25×10^{-3}	1.69×10^{-9}
Medical and industrial isotope glovebox explosion	5.00×10^{-8}	1.48×10^{-3}	1.92×10^{-6}
Beyond-design-basis earthquake	8.25×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	1.94×10^{-7}	5.94×10^{-3}	1.19×10^{-5}
35-year FMEF risk summation	6.79×10^{-6}	2.08×10^{-1}	4.17×10^{-4}
35-year risk summation for Option 3	$\mathbf{6.80\times10^{-6}}$	$\mathbf{2.09\times10^{-1}}$	$\mathbf{4.17\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–59 Risk Summation for Alternative 1—Option 4

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 4	**4.51×10^{-4}**	**5.35×10^{-1}**	**3.50×10^{-4}**

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–60 Risk Summation for Alternative 1—Option 5

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and Industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 5	**4.51×10^{-4}**	**4.07×10^{-1}**	**3.50×10^{-4}**

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–61 Risk Summation for Alternative 1—Option 6

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
FMEF (full processing)			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotope localized solvent fire	6.13×10^{-8}	1.25×10^{-3}	1.69×10^{-9}
Medical and industrial isotope glovebox explosion	5.00×10^{-8}	1.48×10^{-3}	1.92×10^{-6}
Beyond-design-basis earthquake	8.25×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	1.94×10^{-7}	5.94×10^{-3}	1.19×10^{-5}
35-year FMEF risk summation	6.79×10^{-6}	2.08×10^{-1}	4.17×10^{-4}
35-year risk summation for Option 6	$\mathbf{6.80\times10^{-6}}$	$\mathbf{2.09\times10^{-1}}$	$\mathbf{4.17\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–62 Risk Summation for Alternative 2—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target Dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.58\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–63 Risk Summation for Alternative 2—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	$\mathbf{1.51\times10^{-5}}$	$\mathbf{3.01\times10^{-2}}$	$\mathbf{3.53\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–64 Risk Summation for Alternative 2—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	$\mathbf{2.88\times10^{-6}}$	$\mathbf{1.14\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–65 Risk Summation for Alternative 2—Option 4

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 4	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.60\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–66 Risk Summation for Alternative 2—Option 5

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 5	$\mathbf{1.49\times10^{-5}}$	$\mathbf{3.18\times10^{-2}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–67 Risk Summation for Alternative 2—Option 6

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 6	$\mathbf{2.88\times10^{-6}}$	$\mathbf{1.15\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–68 Risk Summation for Alternative 2—Option 7

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 7	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.57\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–69 Risk Summation for Alternative 2—Option 8

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 8	$\mathbf{1.50\times10^{-5}}$	$\mathbf{2.95\times10^{-2}}$	$\mathbf{3.52\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–70 Risk Summation for Alternative 2—Option 9

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 9	**2.88×10^{-6}**	**1.13×10^{-1}**	**3.50×10^{-4}**

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–71 Risk Summation for Alternative 3—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{9.18\times10^{-5}}$	$\mathbf{2.19\times10^{-1}}$	$\mathbf{5.00\times10^{-4}}$

Table I–72 Risk Summation for Alternative 3—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	$\mathbf{4.95\times10^{-5}}$	$\mathbf{9.11\times10^{-2}}$	$\mathbf{5.00\times10^{-4}}$

Table I–73 Risk Summation for Alternative 3—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	**3.76×10^{-5}**	**1.75×10^{-1}**	**5.00×10^{-4}**

Table I–74 Risk Summation for Alternative 4—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{8.98\times10^{-5}}$	$\mathbf{2.13\times10^{-1}}$	$\mathbf{4.49\times10^{-4}}$

Table I–75 Risk Summation for Alternative 4—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	**4.75×10^{-5}**	**8.48×10^{-2}**	**4.49×10^{-4}**

Table I–76 Risk Summation for Alternative 4—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	$\mathbf{3.55\times10^{-5}}$	$\mathbf{1.68\times10^{-1}}$	$\mathbf{4.49\times10^{-4}}$

I.2 HAZARDOUS CHEMICAL ACCIDENT IMPACTS ON HUMAN HEALTH

I.2.1 Irradiation Facility

I.2.1.1 Advanced Test Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at ATR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 targets for plutonium-238 production are postulated at ATR.

I.2.1.2 High Flux Isotope Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at HFIR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 targets for plutonium-238 production are postulated at HFIR.

I.2.1.3 Commercial Light Water Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at the generic CLWR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 for plutonium-238 production are postulated at the generic CLWR.

I.2.1.4 Fast Flux Test Facility

The FFTF primary heat transport system contains a substantial quantity of liquid sodium. In the event of an accident, sodium could be released from the primary system. Sodium mainly forms sodium oxide upon burning; however, in the present of moisture, it can form the more hazardous sodium hydroxide. The beyond-design-basis core melt accident would result in the greatest sodium release to the environment. Previous analyses have shown that the concentration of sodium hydroxide released in this accident scenario would be below the Emergency Response Planning Guidelines-1 (ERPG-1) limit of 2 milligrams per cubic meter at 100 meters (DOE 1997). Since this is the bounding sodium release and all the sodium released was assumed to be in the form of sodium hydroxide, no further analysis was performed for this NI PEIS.

I.2.2 Processing Facility

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, and postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, does not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

No chemical processing activities are currently performed at FDPF and FMEF, and no chemicals are stored in these facilities. If either of these facilities is selected to support the plutonium-238 production program, a hazardous chemical accident analysis will be required. The analysis for FDPF and FMEF assumes that the chemical inventory required for 1 year of operation is stored in the facility and that each chemical is stored in a single tank or container with no mitigating design features (e.g., dikes to limit the spill area).

I.2.2.1 Accident Scenario Selection

This section describes the process used to identify the chemicals for the accident analysis, the methodology used in analyzing potential accidents involving hazardous chemicals, the baseline accident scenarios, and the potential health risks associated with a release from identified scenarios. The anticipated chemical inventory for 1 year of plutonium-238 processing at FMEF and FDPF is given in **Table I–77**.

Table I–77 Anticipated Annual Inventory for Plutonium-238 Processing

Chemical	Inventory (pounds)	TPQ (pounds)	Chemical	Inventory (pounds)	TPQ (pounds)
Aluminum nitrate	5.2	Not in list	Acetone	69	–
Aluminum powder	303	–	Acetylene	16	–
Aluminum stearate	0.6	Not in list	Adogen 364	459	Not in list
Anion exchange resin	97	Not in list	Argon	1,333	Not in list
Ascorbic acid	157	Not in list	Compressed air	135	Not in list
Diethyl benzene	485	Not in list	Devcon 5-minute epoxy resin	7.4	Not in list
Dodecanol	93	Not in list	Helium	62	Not in list
Ferrous sulfamate	7	Not in list	Hydrochloric acid	321	–
Hydrazine nitrate	28	Not in list	Hydrogen (2-5 percent) in argon	24	Not in list
Hydroxylamine nitrate	60	Not in list	Hydrogen peroxide solution (< 52 percent)	8.8	1,000 lb for >52 percent
Methanol	17	Not in list	Nitric oxide	156	100
Nitric acid	2,170	1,000	Nitrogen	833	Not in list
Normal paraffin hydrocarbons	157	Not in list	Oxygen	29	Not in list
Oxalic acid	56	Not in list	P-10 nuclear counter mixture	1,184	Not in list
Polystyrene resin	783	Not in list	Potassium carbonate	5.5	Not in list
Sodium fluoride	0.6	–	Propane	450	–
Sodium hydroxide	1,078	–	Sodium carbonate	8.8	Not in list
Sodium nitrate	1,146	Not in list	Sodium hydroxide (40-50 percent)	7,422	–
Sodium nitrite	1.7	–	Sodium hydroxide (>10 percent solution)	1,068	–
Tributyl phosphate	849	Not in list	Sodium hypochlorite solution	27	–

Key: TPQ, Extremely Hazardous Substances List Threshold Planning Quantity Value; <, less than; >, greater than; –, no value in the list.
Source: EPA 1998.

Only the anticipated annual usage of nitric acid and nitric oxide for plutonium-238 processing exceeds the Threshold Planning Quantities for these substances as stipulated on the Extremely Hazardous Substances List provided in Section 3.02 of the Emergency Planning and Community Right-to-Know Act (EPA 1998). The respective Threshold Planning Quantities for nitric acid and nitric oxide are 454 kilograms (1,000 pounds) and 45.4 kilograms (100 pounds). Since inventories of these chemicals exceed the Threshold Planning Quantities, an evaluation of potential accident scenarios is presented in this NI PEIS.

I.2.2.2 Accident Scenario Descriptions

Two accidental chemical scenarios are postulated for this NI PEIS: the accidental uncontrolled release of nitric acid, and the accidental uncontrolled release of nitric oxide.

I.2.2.2.1 Nitric Acid Release

The nitric acid release scenario was developed on the basis of the following assumptions: the nitric acid released from the tank is red fuming nitric acid (100 percent). A catastrophic tank failure is the initiating event. There are no engineered safety features for the tank. The tank is in an unsheltered building in an open rural area. The release fraction is 100 percent. However, the actual amount of nitric acid that volatilizes to the atmosphere was determined by the method described in the *Technical Guidance for Hazards Analysis* (EPA 1987).

The consequences of the postulated nitric acid release scenario are overstated because of the conservatism of two assumptions: 100 percent red fuming nitric acid and a lack of engineered safety features to restrict releases from spills. Facilities under consideration for this program do not permit storage of red fuming nitric acid, which has a high vapor pressure, and they have engineered safety features (e.g., sloped floors and dikes) to restrict releases from spills.

I.2.2.2.2 Nitric Oxide Release

The analysis postulated a storage cylinder failure. A release fraction (percentage of material released) of 100 percent was used. An aggregated release of 71.8 kilograms (158 pounds) for nitric oxide gas was postulated. The rate of release of nitric oxide was calculated by the method described in the *Technical Guidance for Hazards Analysis* (EPA 1987).

I.2.2.3 Hazardous Chemical Accident Analysis Methodology

The potential health impacts of accidental releases of hazardous chemicals were assessed by comparing estimated airborne concentrations of the chemicals with the ERPG developed by the American Industrial Hygiene Association. The ERPG values are not regulatory exposure guidelines and do not incorporate the safety factors normally included in healthy worker exposure guidelines. ERPG-1 values are maximum airborne concentrations below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health impacts. ERPG-2 values are indicative of irreversible or serious health effects or impairment of an individual's ability to take protective action. ERPG-3 values are indicative of potentially life-threatening health effects.

No approved ERPG levels are available for nitrous oxide (Kelly 1999). The ERPG values for nitric acid are presented in **Table I–78**. The ERPG values referenced by Kelly (1999) were used in this NI PEIS.

Table I–78 Emergency Response Planning Guideline Values for Nitric Acid

ERPG Level	DOE 1997	Kelly 1999
ERPG-1	2 parts per million	0.5 parts per million
ERPG-2	15 parts per million	10 parts per million
ERPG-3	30 parts per million	25 parts per million

Key: ERPG, Emergency Response Planning Guideline.

There are also no ERPG values for nitric oxide. For these cases, the "level of concern" has been estimated by using one-tenth of the "immediately dangerous to life and health" level for that substance—i.e., 100 parts per million—as published by the National Institute for Occupational Safety and Health (NIOSH 1997). For this NI PEIS, therefore, the level of concern for nitric oxide is 10 parts per million. Level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects or death as a result of a single exposure for a rather short period of time (EPA 1987).

I.2.2.3.1 Receptor Description

The potential health impacts of the accidental release of nitric acid and nitric oxide were assessed for three types of receptors:

1. A noninvolved worker—a worker assumed to be 640 meters (0.4 mile) from the point of release.

2. Site boundary maximally exposed individual—a hypothetical member of the public off site at the nearest point of access (7,210 meters [4.5 miles] for FMEF and 13,952 meters [8.7 miles] for FDPF) to the point of release.

3. Nearest highway maximally exposed individual—a hypothetical member of the public at the nearest point on an onsite highway (7,100 meters [4.4 miles] for FMEF and 5,800 meters [3.6 miles] for FDPF) to the point of release.

Facility workers (i.e., those individuals in the building at the time of the accident) are assumed to be killed by the release.

I.2.2.3.2 Analysis Computer Code

The computer code used for estimation of airborne concentrations was the Computer Aided Management of Emergency Operations Areal Locations of Hazardous Atmospheres (ALOHA), developed by the National Safety Council, EPA, and the National Oceanic and Atmospheric Administration.

DESCRIPTION OF THE MODEL

The atmospheric dispersion modeling for the above scenarios was conducted using the ALOHA 5.05 computer code (NSC 1990). The ALOHA code was designed for use by first responders. The model is most useful for estimating plume extent and concentration downwind from the release source for short-duration chemical accidents. It uses a Gaussian dispersion model to describe the movement and spreading of a gas that is neutrally buoyant. For heavier-than-air vapor releases, the model uses the same calculations as those used in EPA's Dense Gas Dispersion Model DEGADIS 2.1 (EPA-450/4-89-019). There are a number of limitations to the model:

1. ALOHA is not intended for modeling accidents involving radioactive chemicals.

2. It is not intended for use in modeling the permitting of stack gas or chronic, low-level (fugitive) emissions.

3. The ALOHA-DEGADIS heavy-gas module is more conservative than the DEGADIS model, which could result in a larger footprint than would actually be expected.

4. ALOHA does not consider the effects of thermal energy from fire scenarios or the byproducts of chemical reactions.

5. ALOHA does not include the process needed to model particulate dispersion.

6. ALOHA does not consider the shape of the ground under the spill or in the area affected by the plume.

7. ALOHA does not estimate concentrations under very low wind speeds (less than 1 meter [3.3 feet] per second), as the wind direction may become inconsistent under these conditions.

8. For very stable atmospheric conditions (usually late night or early morning), there will be uncertainties in the model estimates due to shifting wind directions and virtually no mixing of the plume into the surrounding air. The estimates may in fact, reflect to high airborne concentrations for long periods of time or at great distances from the release source.

9. ALOHA does not accurately represent variations associated with near-field (close to the release source) patchiness. In the case of a neutrally buoyant gas, the plume will move downwind; very near the source, however, it can be oriented in a different direction (e.g., going backward) due to the effect of drifting eddies in the wind.

WEATHER CONDITION ASSUMPTIONS

The model results are presented for atmospheric Stability Classes D and F, with wind speeds of 5.3 meters (17 feet) per second and 1.5 meters (4.9 feet) per second, respectively. Atmospheric Stability Class D is considered to be representative of average weather conditions; Stability Class F is considered to be representative of worst-case weather conditions. These weather conditions were selected because they are recommended in *Technical Guidance for Hazards Analysis* (EPA 1987). **Table I–79** presents the model parameter values for these weather conditions.

Table I–79 Analysis Weather Conditions

Parameter	Average Condition Stability Class D	Worst-Case Condition Stability Class F
Ambient air temperature	75 °F	60 °F
Relative humidity	50 percent	25 percent
Cloud cover	50 percent	20 percent
Average wind speed	5.3 meters per second	1.5 meters per second

Source: EPA 1987.

I.2.3 Human Health Impacts

The potential health impacts of the accidental releases were assessed by comparing the modeled ambient concentrations of nitric acid and nitrous oxide at each of the previously identified receptor locations with the ERPGs. The estimated airborne concentrations of nitric acid and nitric oxide are presented in **Table I–80** and **Table I–81**, respectively. **Table I–82** and **Table I–83** present of the impacts data for nitric acid and nitric oxide.

Table I–80 Airborne Concentration Estimates for Nitric Acid Release Scenarios

Downwind Distance from Source (meters)	Nitric Acid Concentration Under Stability Class D (parts per million)		Nitric Acid Concentration Under Stability Class F (parts per million)	
Facility	**FMEF**	**FDPF**	**FMEF**	**FDPF**
30	1,200	1,130	1,070	1,040
640	3.3	3.3	8.6	8.4
1,000	1.4	1.4	3.9	3.9
3,000	0.17	0.17	0.5	0.5
5,000	0.06	0.06	0.2	0.2
Nearest highway[a]	0.03	0.05	0.1	0.15
Site boundary [b]	0.03	(c)	0.1	(c)

a. FMEF = 7,100 meters (4.4 miles); FDPF = 5,800 meters (3.6 miles).
b. FMEF = 7,210 meters (4.5 miles); FDPF = 13,952 meters (8.7 miles).
c. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Source: Calculated results.

Table I–81 Airborne Concentration Estimates for Nitric Oxide Scenarios

Downwind Distance from Source (meters)	Nitric Oxide Concentration Under Stability Class D (parts per million)		Nitric Oxide Concentration Under Stability Class F (parts per million)	
Facility	**FMEF**	**FDPF**	**FMEF**	**FDPF**
30	1,370	1,370	9,990	9,480
640	4.2	4.2	66	67.5
1,000	2	2	29.2	29.6
3,000	0.36	0.36	3.6	3.6
5,000	0.17	0.17	1.2	1.2
Nearest highway[a]	0.09	0.09	0.55	0.87
Site boundary [b]	0.09	(c)	0.53	(c)

a. FMEF = 7,100 meters (4.4 miles); FDPF = 5,800 meters (3.6 miles).
b. FMEF = 7,210 meters (4.5 miles); FDPF = 13,952 meters (8.7 miles).
c. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Source: Calculated results.

Table I–82 Summary of Impacts Data for Release Scenarios (Nitric Acid)

Evaluation Parameter		FMEF (Stability Class D)	FDPF (Stability Class D)	FMEF (Stability Class F)	FDPF (Stability Class F)
Maximum distance to (meters)	ERPG-1 ERPG-2 ERPG-3	2,000 500 375	2,000 500 375	3,000 600 450	3,000 600 450
Noninvolved worker (640 meters)	Parts per million (ppm) Level of concern Potential health effects	3.3 < ERPG-2 Mild, transient	3.3 < ERPG-2 Mild, transient	8.6 < ERPG-2 Mild, transient	8.4 < ERPG-2 Mild, transient
Nearest highway maximally exposed individual	Parts per million (ppm) Level of concern	0.03 < ERPG-1	0.05 < ERPG-1	0.1 ERPG-1	0.15 ERPG-1
Site boundary maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.03 < ERPG-1 None	(a) < ERPG-1 None	0.1 ERPG-1 Mild, transient	(a) ERPG-1 Mild, transient

a. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Key: <, less than; ERPG, Emergency Response Planning Guideline.

Table I–83 Summary of Impacts Data for Release Scenarios (Nitric Oxide)

Evaluation Parameter		FMEF (Stability Class D)	FDPF (Stability Class D)	FMEF (Stability Class F)	FDPF (Stability Class F)
Maximum distance (meters)	To concentrations of level of concern	500	500	1,900	2,000
Noninvolved worker (640 meters)	Parts per million (ppm) Level of concern Potential health effects	4.2 < LOC Mild, transient	4.2 < LOC Mild, transient	66 > LOC Serious	67.5 > LOC Serious
Nearest highway maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.09 < LOC None	0.09 < LOC None	0.55 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.09 < LOC None	(a) < LOC None	0.53 < LOC None	(a) < LOC None

a. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Key: <, less than; >, greater than.

I.2.3.1 Impacts to Noninvolved Workers

Nitric Acid. A noninvolved worker is assumed to be located 640 meters (0.4 mile) from the point of release. The concentrations of nitric at that distance range from 3.3 to 8.6 parts per million for FMEF and 3.3 to 8.4 parts per million for FDPF, given assumed meteorological conditions. The maximum estimated airborne concentration at 640 meters (0.4 mile) Stability Class F exceeds the ERPG-1 value of 0.5 part per million for nitric acid, which suggests the potential for only mild, transient, and reversible health impacts on a noninvolved workers at that distance from the release.

Nitric Oxide. For the nitric oxide release scenarios, the concentrations at 640 meters (0.4 miles) range from 4.2 to 66 parts per million for FMEF and 4.2 to 67.5 parts per million for FDPF, given assumed meteorological conditions. As a result, the maximum estimated airborne concentration at 640 meters (0.4 miles) exceeds the level-of-concern value of 10 parts per million for nitric oxide, which suggests that a noninvolved worker may experience irreversible or serious, but not life-threatening, health impacts if the exposures are not mitigated.

I.2.3.2 Impacts on Access Roads

Nitric Acid. The receptor at the nearest highway is assumed to be located 7,100 meters (4.4 miles) and 5,800 meters (3.6 miles) for FDPF from the points of release at FMEF and FDPF respectively. For the nitric acid release scenarios, the receptor on the nearest highway could be exposed to a nitric acid concentration of 0.03 to 0.05 part per million under Stability Class D conditions, which is below the ERPG-1 value for nitric acid of 0.5 part per million. Exposures to concentrations below the ERPG-1 value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the offsite receptor may be exposed to a nitric acid concentration of about 0.1 to 0.15 part per million, which is below the ERPG-1 value for nitric acid of 10 parts per million. Exposure of the receptor to concentrations greater than the ERPG-1 value may have only mild, transient and reversible health impacts.

Nitric Oxide. For the nitric oxide release scenarios, the receptor on the nearest highway could be exposed to concentrations of 0.09 part per million under Stability Class D conditions, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposures to concentrations below the level-of-concern value are not expected to produce any adverse health effects for the receptor. Under Stability Class F conditions, the offsite receptor may be exposed to a nitric oxide concentration of about 0.55 to 0.87 parts per million, which is below the level-of-concern value for nitric oxide of 10 parts per million.

I.2.3.3 Offsite Impacts

The site boundary receptor is assumed to be located at a distance of 7,210 meters (4.5 miles) and 13,952 meters (8.7 miles) from the points of release at FMEF and FDPF, respectively. ALOHA does not draw any plume larger than 10 kilometers (6.2 miles) (NSC 1990). The FDPF site boundary is 13.3 kilometers (8.3 miles) from the point of release. Therefore, no impacts to site boundary receptor at FDPF were performed. Health impacts on the nearest highway at a distance of 5.8 kilometers (3.6 miles) from the point of release were only mild, transient, and reversible. Exposure of the receptor at concentrations below the level-of-concern value may have only mild, transient, and reversible health impacts. At a distance of 14.0 kilometers (8.7 miles) from the point of release, adverse health effects are not expected.

Nitric Acid. For the nitric acid release scenarios, the site boundary receptor at Hanford could be exposed to a nitric acid concentration of 0.03 part per million under Stability Class D conditions, which is below the ERPG-1 value for nitric acid of 0.5 part per million. Exposures to concentrations below the ERPG-1 value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the site boundary receptor may be exposed to a nitric acid concentration of about 0.1 part per million, which is below the ERPG-1 value for nitric acid of 10 parts per million. Exposure of the receptor at concentrations greater than the ERPG-1 value may have only mild, transient, and reversible health impacts.

Nitric Oxide. For the nitric oxide release scenarios, the site boundary receptor exposures were 0.09 part per million under Stability Class D conditions, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposures to concentrations below the level of concern value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the site boundary receptor may be exposed to a nitric oxide concentration of about 0.53 part per million, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposure of the receptor at concentrations below the level-of-concern value may produce only mild, transient, and reversible health impacts.

I.2.3.4 Uncertainties

This screening-level analysis is subject to a number of uncertainties relative to the atmospheric dispersion modeling. Among those uncertainties are the following:

1. On the day of an accident, it will undoubtedly be very difficult to establish exactly the rate or magnitude of the release.

2. The weather conditions and wind speed may well be different from those used in the analysis.

3. The dispersion modeling does not take into account the deposition of highly reactive vapors onto surfaces, including equipment, groundwater, and vegetation. This means that the model overestimates airborne concentrations at longer distances.

4. Overall, the uncertainties in predicted airborne concentrations may be as large as a factor of plus or minus two times the estimated concentration.

5. In view of these uncertainties, the results of this analysis should be considered only as screening-level estimations.

I.3 INDUSTRIAL SAFETY

Estimates of potential industrial impacts to workers during construction, irradiation, fabrication, and processing were evaluated based on DOE and Bureau of Labor Statistics data. Impacts are classified into two groups: total recordable cases and fatalities. A recordable case includes work-related death, illness, or injury that resulted in loss of consciousness, restriction of work or motion, transfer to another job, or required medical treatment beyond first aid.

DOE and contractor total recordable cases and fatality incidence rates were obtained from the CAIRS database (DOE 2000a, 2000b). The CAIRS database is used to collect and analyze DOE and DOE contractor reports of injuries, illnesses, and other accidents that occur during DOE operations. The 5-year average (1995–1999) rates were determined for average construction total recordable cases, average operation total recordable cases, and average operation fatalities. The average construction fatality rate was obtained from the Bureau of Labor Statistics (Toscano and Windau 1998).

Table I–84 presents the average occupational total recordable cases and fatality rates for construction and operation activities.

Table I–84 Average Occupational Total Recordable Cases and Fatality Rates (per worker-year)

Labor Category	Total Recordable Cases	Fatalities
Construction	0.053	0.000139
Operation	0.033	0.000013

Expected impacts (both annual and for the duration of the activity) to workers at each facility for construction and operation are presented in **Table I–85**.

Table I–85 Industrial Safety Impacts from Construction and Operation

Facility	Estimated Number of Workers	Construction or Operation Duration (years)	Annual Total Recordable Cases	Activity Duration Total Recordable Cases	Annual Fatalities	Activity Duration Fatalities
Construction						
Low-energy accelerator	75	3	4.0	12	0.010	0.03
High-energy accelerator	410	5	22	110	0.057	0.285
New research reactor	160	7	8.5	59.5	0.022	0.154
Operation						
ATR[a]	0	35	NA	NA	NA	NA
HFIR[a]	0	35	NA	NA	NA	NA
CLWR[a]	0	35	NA	NA	NA	NA
FFTF	242	35	8.0	280	0.0031	0.109
Low-energy accelerator	13	35	0.4	14	0.00017	0.00595
High-energy accelerator	225	35	7.4	259	0.0029	0.102
New research reactor	120	35	4.0	140	0.0016	0.056
REDC	116	35	3.8	133	0.0015	0.0525
FDPF	75	35	2.5	87.5	0.00098	0.0343
FMEF	105	35	3.5	123	0.0014	0.049
RPL/306–E	30	35	1.0	35	0.00039	0.0137
New support facility	100	35	3.3	116	0.0013	0.0455

a. Not applicable. No additional workers would be required for the proposed activities evaluated in this NI PEIS.

Computation of total recordable cases and fatalities expected during construction or modification of target fabrication and processing facilities, prior to operation, have been neglected because of the relatively short duration of these activities.

As expected, the incidence of impacts, above and beyond those requiring first aid, do indeed exceed impacts from radiation and hazardous chemical accidents evaluated in this NI PEIS. No fatalities would be expected from either construction or operation of any facility.

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